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Detailed analyses and substantial structures are required in Light Water Reactor power plants, to protect against the damaging effects of pipe breaks. As an alternative, one might show that the growth of any crack in a particular pipe would lead to fluid leaks which would be detected long before such a crack would result in a pipe break. This principle is referred to as "Leak-Before-Break".
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This thesis presents the results of a study which was conducted to assess the potential impact of Leak-Before-Break on the design of modern Light Water Reactors. It was determined that a majority of pipe rupture restraints could be eliminated in a typical plant, with a potential cost savings of tens of millions of dollars per plant. Assumptions about reactor operating conditions and leak detection capability are critical.

Some recommendations made in this thesis are: safety margins to be used in leak-before-break determination should be based on actual risk; and requirements for leak detection sensitivity in nuclear plants should be based on specific needs.

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LIGHT WATER REACTOR DESIGN

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POTENTIAL EFFECTS OF LEAK-BEFORE-BREAK ON LIGHT WATER REACTOR DESIGN

by

Paul E. Roege

Submitted to the Department of Nuclear Engineering on the 9th of August 1985, in partial fulfillment of the requirements for the Degrees of Nuclear Engineer and Master of Science in Nuclear Engineering.

ABSTRACT

Leak-before-break is a concept which has been proposed as an alternative to postulating pipe rupture in light water reactors. The principle requires that pipes be able to withstand cracks which are large enough to leak at a rate which would be detected by installed leak detection instrumentation. The concept has been applied to primary coolant loops in pressurized water reactors, but has not been extended to include other pipe systems.

The potential for applying leak-before-break to other systems was studied, along with methods for demonstrating leak-before-break. The results were applied to the design of a light water reactor power plant.

For most pipes of concern, leak-before-break could apply. Only the smallest pipes, primarily those under 8 inches (21 cm) in diameter, lack potential under this concept. A large number of protective measures taken against pipe rupture could be eliminated. The effect will be greatest on future plants, where pipe rupture protection has not yet been installed, nor planned for.

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Thanks to my advisors, particularly to Professor Eric Beckjord. He gave generously of his time and energy to provide guidance and encouragement. The leak-before-break project, of which this thesis is a part, has been accomplished in no small part due to his leadership.

Finally, thanks to my wife, Sherry, and my daughters, Carol and Diane, whose cooperation and understanding have allowed me to devote many hours to this study, often allowing them less attention then they deserved.

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Chapter 1

INTRODUCTION TO LEAK-BEFORE-BREAK

1.1. Introduction

When designing nuclear power plants, attention must be given to protecting systems which contribute to the safety of the plant. In fact, any system which is considered to be necessary for the safe shutdown of the plant is referred to as an essential safety system, and must be protected from potentially damaging effects. A break in a high energy pipe could result in pipe whip and high velocity fluid jets, which could create significant damage. Consequently, any pipe larger than 1 inch (2.5 cm) nominal size, which carries high energy fluid during normal operation, must be evaluated for the damage it could cause, should it break. Appropriate measures must be taken in the design process to protect essential safety systems from the damaging effects of such a break. One measure which is taken to that end is the addition of sometimes massive structures, called pipe rupture restraints, which would restrain the motion of pipe ends in the event of a break. Figure 1.1. shows a typical pipe restraint with supporting framework.

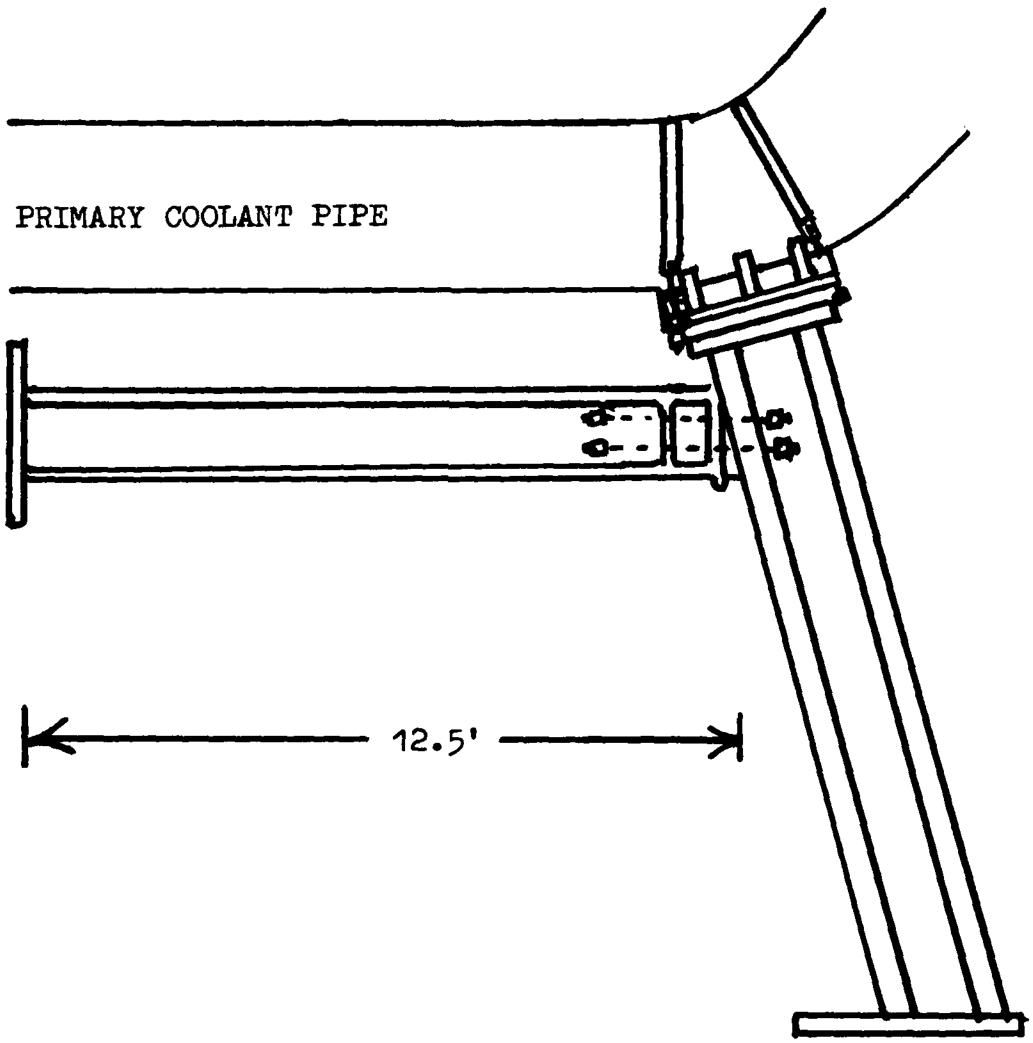

Figure 1.1. Typical Pipe Restraint and Associated Supporting Framework

Another measure is the construction of large shields which protect such safety systems from the effects of fluid jets which would come from a break or large crack in a pipe. These structures add substantially to the costs of nuclear power, while their actual contributions to plant safety are questionable.

Although many in the nuclear industry have long suspected that the pipes used in nuclear plants were very unlikely to break, the existing analytical arguments were not convincing enough to justify removing the requirement for the pipe restraints and jet impingement shields. Despite the fact that reactor piping is required to undergo before and in-service inspection, there was fear that some undetected flaw could grow into a break. Recent advances in the fields of cracking, or fracture mechanics, and two-phase flow allow reliable calculations which may show that any crack in a given pipe would leak so much that it would be detected by installed leak detection instrumentation before resulting in a break. The concept of avoiding pipe breaks by using leak detection is called "leak-before-break".

The Nuclear Regulatory Commission has recently accepted a leak-before-break arguement to justify the removal of pipe restraints on large primary coolant pipes in some plants. Because of the sheer size and crack resistance of the primary coolant pipes, it has been shown that a crack would have to become very large before developing into a break. Leak rates of tens of gallons per minute would be expected from a crack of one half the size which would result in a pipe break. The operating specifications for present pressurized water reactors call for the ability to detect primary coolant water leakage at a rate of 1 gallon per minute (3.8 liter/min) within one to four hours, depending on the plant.

While this leak detection capability was judged sufficient to avoid breaks in primary coolant loop piping, it was unclear whether this leak-before-break reasoning could be applied to other pipes. If a similar argument could be applied to other systems in nuclear plants, many of the pipe rupture protection devices could be removed or modified. The result would be substantial savings in plant costs, reliability, and operational radiation doses. This paper will describe methods for demonstrating leak-before-break, assess the potential impacts of leak-before-break on the design of light water reactors, and present the issues which should be considered when deciding to what degree to implement leak-before-break. A 4-loop Westinghouse Pressurized Water Reactor (PWR) power plant was chosen as an example plant. The plant identity is withheld at the owners' request.

1.2. Background

When the first nuclear power plants were designed, the pressurized components were required to comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. In fact, section 3 of that code was created for nuclear applications. This code was intended to ensure that the pressure boundaries, made up of pipe and fluid-carrying component walls, would not fail. The major concerns were loss of coolant, resulting in radiation exposure and possible reactor core damage, and possibly injuries associated with bursting pipes or other components.

Another requirement placed on nuclear reactor design was to consider the potential effects of fluid jets and pipe whip associated with pipe breaks, on systems required for the safe shutdown of the plant. Although this requirement, as set forth in General Design Criterion 4 of Appendix A to 10 CFR part 50, existed before 1972, it had little effect on plant design. The regulation did not give specific guidelines on when to worry about these breaks, nor what measures to take in order to protect against them.

In December 1972, guidance on how to meet this requirement was set forth in the "Giambusso Letter", reference 13. The letter defined which pipes should be considered potential threats, stated that designers should postulate that they would break at certain locations, defined the locations where breaks should be postulated, then described appropriate measures for protecting essential systems from such breaks. Since December, 1972, when the Giambusso letter was distributed, substantial resources have been expended to postulate break locations in pipes, analyze the potential for damage to safety systems, and install protective devices in order to minimize such damage.

At the same time, evidence has grown that suggests that the probability of complete breaks in reactor coolant piping is particularly small, and that any crack which approached breaking size would be found by installed leak detection systems.

Research conducted at Lawrence Livermore Laboratory, as cited in reference 18, demonstrated the very low probabilities of double-ended guillotine breaks, or DEGB in primary coolant piping in pressurized water reactors. As an example, for Westinghouse Pressurized Water Reactor plants east of the Rocky Mountains, they calculated that the probability of break in primary coolant piping due to crack growth is on the order of 10^{-12} per year, while the probability of detecting a leak due to a growing crack is of order 10^{-7} per year. This suggested that the probability of break was very low, and that 100,000 leaks would be found before a crack would result in a break.

The Livermore study showed not only that breaks due to crack growth were highly unlikely, but that leak detection was a reliable warning system in avoiding such breaks. Based on the results of this study, and the recommendations of experts in the fields of fracture mechanics and fluid flow, the U.S. Nuclear Regulatory Commission published preliminary guidelines for satisfying leak-before-break criteria in Reference 18. Several studies are now being conducted by government agencies and by private industry to evaluate these guidelines, and the likely impact that leak-before-break might have on existing and future nuclear power plants.

1.3. Objectives

For leak-before-break to be valuable, it must provide an avenue toward improving some aspect of nuclear plants. Improvements might be possible in the form of cost savings, improved reliability, or reduced radiation doses for workers or the public. For a nuclear plant, the most desireable improvements would be simplification of design or operation, reductions in capital cost, increased plant availability and reliability, and reduced radiation dose rates.

Nuclear power plants are among the most complex construction projects of modern times. As a result, construction times are long, and further delays result from unforseen complications, such as modifications to the design, which must be made when there are unforseen interferences between components, or when it is found that one activity cannot proceed before another incomplete one is done.

Capital costs constitute a larger portion of the cost of electricity from nuclear plants than they do in the case of other types of generating plants. Further, capital costs have recently become so high as to become prohibitive. In some cases, this has forced utilities to convert a partially constructed nuclear plant to a fossil fueled plant, or has made financing completely impossible, so that a nuclear plant under construction must be cancelled.

Increased availability and reliability are two related concerns which are both related to the chances of an accident. Nuclear plants in America have experienced decreasing availability rates over the last several years, while reactors in other countries have operated for much higher percentages of the time. Apparently, it is more common for U.S. plants to shut down, either because of a perceived danger, or because of a need for maintenance. Each reactor shutdown adds significantly to the cost of electricity, while the cause of the shutdown may be a malfunction which increases the risk of accident. Everyone in the nuclear industry has a responsibility to protect the public from the risks of accidents.

Finally, the risk of radiation exposure is a concern which is hard to compare to others. Because radiation doses are hard to measure accurately, and the effects are not as well understood as, say, financial concerns, a reduction in expected dose rates would be a very compelling objective for any improvement to nuclear plants. Leak-before-break would be most valuable if it can offer advancements toward the objectives of: reduced operational dose rates, improved plant reliability and availability, and reduced complexity and capital costs for the plant.

The major contributions that the work represented by this thesis has made toward leak-before-break are depicted in Table 1.1.

Table 1.1

CONTRIBUTIONS OF THIS WORK

-Clarifies the Requirements for Applying
Leak-Before-Break

-Identifies Appropriate Models for
Leak-Before-Break Analysis

-Evaluates Relationships Between
*Pipe Size
*Applied Stress
and Leak Rate

-Examines Application of Leak-Before-Break
to Specific Plant Area

-Establishes Probable Degree to Which
Leak-Before-Break Will be Applied

-Recommends Modifications to Leak-Before-Break
Requirements and to Plant Design Philosophy

Chapter 2

PRINCIPLES OF LEAK-BEFORE-BREAK

2.1 Introduction

Leak-before-break is the principle of avoiding pipe breaks by using installed leak detection capability to detect cracks which could lead to breaks. In concert with previous philosophy in the nuclear industry, it is important to provide a defense in depth against public and occupational exposure to radiation. The principles for applying leak-before-break are based on this philosophy. They include ensuring that the chance of cracks is low, then making sure that any crack which should develop would be detected and recognized, resulting in a plant shutdown.

2.2 Philosophy

2.2.1. Defense in Depth

Leak-before-break is based on principles which ensure that the nuclear plant under consideration would not experience a significant accident as a result of pipe break. Elements which contribute to this include very low probability that undetected flaws would grow appreciably, and very high probability that any flaw which did appear would be

Table 2.1

DEFENSE IN DEPTH AGAINST PIPE RUPTURE

-LOW BREAK PROBABILITY

*Quality control of materials and fabrication

*Tough pipe materials

*Flexible pipe system design

*Regulation of plant operating conditions

-HIGH DETECTION PROBABILITY

*Pre-operational inspection

*Pressure testing

*Reliable leak detection

*In-service inspection

detected before it could result in a break. These two elements comprise a two-tiered defense, characteristic of nuclear facilities. One must be able to demonstrate that both of the above conditions are true in order to eliminate pipe whip from consideration in the reactor design. Table 2.1 summarizes the strategy of defense in depth.

2.2.2. Low Break Probability

Several factors support the element of low break probability; quality control in materials and construction, tough pipe materials, flexibility built into piping systems, and strict regulation of operating conditions allowed in the plant are important ones. Arguments which are based on these factors must also be supported by experience with similar types of systems and components.

Quality control is very strict in nuclear-related fabrication. Components which will be incorporated in a nuclear portion of a plant must undergo tough inspections, and receive a certifying stamp. Records must also be kept concerning the origins and history of all such components. Nevertheless, no quality control scheme is perfect, and two faults of this system are; 1) an excessive amount of records is generated, which may obscure important information, and 2) special component designs are often used in place of off-the-shelf parts. Off-the-shelf components generally have proven performance records, while specially designed components lack the benefit of extensive operating experience.

Toughness, or resistance to cracking, is an important consideration in choosing pipe materials. The important issue, however, is not the toughness, or resistance to cracking, of the nominal pipe material, but the toughness at highly stressed locations, possible degraded material conditions, and flaws due to fabrication processes. The Livermore study used statistical information in dealing with these issues. Their recommendations about specific testing required and safety margins to be used take these factors into account.

Flexibility is recognized as an essential quality in successful piping system design. Flexibility is the ability of a system to comply to applied displacements without creating regions of high stress. The notable features of a flexible piping system are: numerous changes in direction, wide radius turns, and the absence of restraining structures. Designing a flexible system is an art. Restricting the movement of the pipes results in reduced flexibility. The very pipe restraints which have been added to nuclear plants to increase plant safety, may detract from the quality of the design if improper installation or unexpected thermal expansion should lead to contact between the pipe supports and the pipe. Given the complexity of nuclear plant design, and the difficulty of predicting all possible sequences of operating conditions, it is likely that a pipe restraint would exert restraining forces on a pipe during some phase of plant operation.

Finally, operating conditions help to ensure that the chances for pipe break are minimal. In particular, operating specifications may ensure that thermal stresses do not exceed the expected values, that materials are not highly loaded while they are below their nil ductility temperature, and that water chemistry is controlled so that stress corrosion cracking may not occur. Thermal stresses are those which occur due to restrained thermal expansion. The restraint may be externally imposed, or may be internal, due to differential heating of the material. Nil ductility temperature is the point above which a given material is ductile, but below which it is brittle. If materials are too highly loaded below their nil ductility temperature, they crack or break.

If certain chemicals in high concentrations are left in contact with some structural materials, even low levels of stress may result in the phenomenon of intergranular stress corrosion cracking, or IGSCC. This is a microscopic effect of corrosion among the material grain boundaries, promoted by local tensile stress and the presence of corrosive elements, and resulting in cracks at those grain boundaries.

2.2.3. Reliable Detection

Having eliminated systems with a history of cracking, the second line of defense is to detect any flaws which might grow despite the first line of defense. All nuclear piping systems are required to undergo periodic inspection for flaws, beginning with an inspection and pressure test before startup. Such periodic inspection is considered reliable enough to safeguard pressurized systems in many other applications, such as scuba tanks and fossil fueled boilers, and for other applications for which fatigue and cracking are concerns. Aircraft undergo only periodic airframe inspections, whether they are civilian or military. There is no requirement for continuous monitoring in these applications, to ensure that failure is not imminent.

In keeping with the principle of defense in depth, however, a second method is deemed necessary to insure against failure in nuclear applications. If it can be shown that leak detection systems provide a reliable test for growing cracks, then devices which protect against damage due to pipe breaks may be eliminated. An implied part of this leak detection scheme is that any leak which is a precursor to pipe break must signal the plant operator to shut the reactor down. This is an important consideration in leak-before-break because a leak from a pipe crack may be curcial, while a leak from some other source may be of lesser concern.

2.3. Principles of Leak-Before-Break

2.3.1. Cracking History

In principle, demonstrating leak-before-break consists of postulating some small initial flaw in a pipe, assuming it grows slowly until it is detected by installed leak detection devices. It must then be demonstrated that the crack would grow little, or not at all beyond this point. Before even considering a pipe for leak-before-break, one must first eliminate from consideration any pipe or system which is subject to crack-producing mechanisms, such as intergranular stress corrosion cracking, thermal fatigue, or water hammer. Operating history provides the support for this criterion.

Eliminating systems with histories of crack-producing mechanisms automatically achieves the purposes of keeping the probability of pipe break low, and maximizing the validity of fatigue calculations. Any of the mentioned mechanisms can lead to sudden pipe failure, even in locations where stresses do not have a high steady state value. Further, historical studies of pipe breaks (Refs. 12, 18, and 32) indicate that virtually every recorded incidence of pipe break could be attributed to environmental effects, such as those mentioned. Some other break producing effects, such as extreme high temperature, need not be considered because they would not occur in light water reactor plant operation.

In light water reactors, there are only three examples of systems with histories of crack producing mechanisms.

Some boiling water reactors have experienced stress corrosion cracking in recirculating lines which were made of type 304 stainless steel. The cracking occurred in the heat affected zones near welds, and was a characteristic effect of welded zones in the material used, and of the degree of chemistry control which can be maintained in the cooling water in boiling water reactors. In reactors which used this material, the recirculation piping has been inspected, and replaced where required by type 314 stainless steel, which is considered immune to stress corrosion cracking (Ref. 12). The question of stress corrosion cracking in BWR recirculating piping is presently under consideration, but the problem is not expected to recur.

The second example of piping with a history of cracking was feedwater piping in pressurized water reactors. After a crack was found on a nozzle where a feedwater pipe entered a steam generator, further inspection uncovered similar cracks on other feedwater lines in the same plant, and in other plants. It was determined that the cause of the cracking was fatigue due to thermal stress. When unheated (ambient temperature) auxiliary feedwater was added to the systems just below the steam generator, it did not mix immediately with the warm water. Where the auxiliary feedwater cooled the pipe material, it created sharp temperature gradients, resulting in high thermal stresses. After many cycles, fatigue cracking occurred. Such cracking was confined to the single location in the feedwater lines, and not considered a concern for the remaining parts of the feedwater system.

Finally, there were numerous cases of cracking in small (<4" or 10.5 cm diameter) pipes, especially chemical and volume control piping in pressurized water reactors. Two factors are considered to have caused this. First, many small lines are intermittently used, and therefore contain stagnant water for long periods of time. This allows chemical agents such as oxygen and chlorine to build up concentrations in some areas, leading to an increased chance of stress corrosion cracking. Another, more common explaination, was that smaller lines tend to vibrate more when excited by flowing fluid and operating pumps and motors. The vibrations are small displacements imposed on the pipes at high frequency. The result was believed to be high cycle fatigue. Reference 12 noted that there were numerous instances of such cracking in small pipes, but in all cases, the result was leakage. No breaks were found in these lines.

A prerequisite for leak-before-break consideration is the absence of mechanisms which drive cracking. In light water reactors, the only cases where this would be a restriction are the nozzle where feedwater lines enter steam generators on pressurized water reactors, and possibly on piping of less than four inch (10.5 cm) nominal diameter. In other cases, leak-before-break may apply, as long as leak detection and stress criteria can be met.

2.3.2. Stress Analysis

Once a system is exonerated from the cracking history requirement, the next step is to perform the normal stress calculations on the pipes in question. In a plant which has already been designed, this has generally been accomplished already. Any piping system to which leak-before-break might apply is already subject to a requirement to find the stresses in the pipes under all postulated operating conditions. This requirement would not change under leak-before-break, but some of the information from this analysis would be required for the leak-before-break analysis.

A major task which is included in the leak-before-break criteria in Reference 18, is a fatigue analysis. The piping systems in a nuclear plant are divided into three classes; Class I systems carry primary coolant water, which is potentially radioactive, while Class II and III systems carry the other operational water in the plant. Because of the concern about leaks of radioactive water, Class I piping must have a detailed fatigue analysis to ensure that it will not fail due to fatigue. This analysis is expensive and time consuming, but Class I piping comprises only a small minority of plant piping.

Under leak-before-break, one must ensure that any included pipe would not crack due to fatigue. The recommendation made by the Piping Review Committee of the Nuclear Regulatory Commission in Reference 18 is that the same fatigue analysis be required on all systems to which leak-before-break would be applied. There is no published simplified method for this analysis, but individuals at the Stone and Webster Engineering Corporation have suggested a simplified argument based on stress and historical motivations (Ref 8). They argue that, if cracking has not been experienced in the system, and if stresses are generally comparable to or smaller than in some other acceptable system, then a full Class I fatigue analysis is not necessary.

2.3.3. Choosing Locations For Analysis

The principles of leak-before-break allow two distinct approaches toward its applicaton. Traditionally, designers have been required to postulate that high energy pipes would break at points of high stress or fatigue, coincident with the worst materials. Where breaks would lead to a hazard to essential safety systems, some protective measure had to be taken. In existing plants, leak-before-break could be applied to those specific postulated breaks which resulted in pipe restraints, resulting in the potential elimination of those restraints.

For future plants, however, it may be simpler not to postulate breaks at all if leak-before-break can be shown for an entire system. To apply leak-before-break to an entire system, either every point in the system must be analyzed, or important points in the system must be chosen for analysis. An acceptable method would have to be used in order to define the important points for analysis.

Section 4.2 presents a study of the effect of applied stresses on leak rate from a crack. The results from such a study may prove helpful in arguing that a particular condition corresponds to a worst case. Cracks are postulated at locations of highest stresses because they are the places where the pipe is most likely to crack or break. These may not correspond to locations which would leak the most from a critical-sized crack. Further, the effects of fatigue are difficult to account for without performing a fatigue analysis on an entire system.

The usual fatigue usage factor, which represents the amount of fatigue resistance which has been used up, is the result of a detailed fatigue analysis. If one wishes to avoid such an analysis, some simpler justification must be found. The postulation of worst case locations in a system is promising for future plants, but a method for doing so is not presented here.

2.3.4. Postulating Flaws

Regardless of the chosen approach, the substance of leak-before-break analysis is to postulate flaws at the locations chosen for analysis. The size of a crack which could be detected due to its characteristic leakage must be compared to the size of a flaw which could lead to a break. If, after appropriate safety margins are applied to crack sizes and leak rates, the "leakage sized" crack is still smaller than the reduced size crack for rupture consideration, then leak-before-break is assured. Because leak rates are a nonlinear function of postulated crack size, while the nominal stresses on the pipe have been resolved by performing stress analyses, it is most reasonable to find the critical crack size for rupture using nominal stress data, apply a safety margin on size, then calculate the expected leak rate. If the calculated leak rate is high enough, detection is assured.

The critical crack size for pipe rupture is the size from which the crack would grow unstably. Unstable crack growth may result because local stresses around the crack are so high that the crack propagates without any further load. This is referred to as the "tearing instability". Alternatively, the cracked pipe may fail in an entirely plastic manner, such that the pipe experiences gross bending, but the crack does not grow. This is the essence of the "plastic instability". The corresponding approaches for predicting these instabilities are referred to as fracture mechanics and limit-load analysis, respectively.

To ensure leak-before-break, finally, it is necessary to make sure that a leakage sized flaw would not grow significantly. While a flaw may be stable, it may be large enough to grow through fatigue over time. Reference 18 suggests that if the leakage sized flaw is subjected to normal plus safe shutdown earthquake (SSE) loads, it should grow very little or not at all. When subjected to loads of $\sqrt{2}$ times normal plus SSE loads, it should remain stable. This is an additional requirement for fatigue analysis, for which there is no simplified approach. Table 2.2 depicts the sequence for performing leak-before-break analysis.

2.3.5. Leak Detection

Leak-before-break depends on the fluid which escapes from a crack to produce a signal to the plant operator that there is a problem. A crack must emit enough fluid to cause a positive signal to be produced by installed leak detection systems. Inside the containment of each nuclear plant in the United States, there are provisions for the detection of

Table 2.2

SEQUENCE FOR LEAK-BEFORE-BREAK ANALYSIS

1. IDENIFY LOCATION FOR ANALYSIS

   -Specific pipe

   -Location of postulated break resulting in pipe restraint

2. DETERMINE OPERATING CONDITIONS

   -Water temperature and pressure

   -Pipe stress state

3. SCREEN FOR CRACKING HISTORY

   -Intergranular Stress Corrosion Cracking

   -Fatigue

   -Water hammer

4. POSTULATE CRACK OF CRITICAL SIZE

   -Tearing instability

   -Plastic instability

5. CHECK REDUCED SIZED CRACK

   (reduced by safety margin)

-leak rate (SHEM, etc.)

-crack growth (da/dN)

-criticality under $\sqrt{2}$ x SSE loads

6. TESTS

-leak rate/10 detectable

-f(da/dN) << pipe diameter

-crack not critical under increased loading

THEN: OK FOR LEAK-BEFORE-BREAK

primary coolant leakage. Such provisions are required by 10 CFR, Part 100. Specific guidance from Regulatory Guide 1.45 recommends that leak detection be provided which can detect leaks of one gallon (3.8 liters) per minute within one to four hours, and several distinct methods should be used. The operating specifications for each plant specify the required leak detection sensitivity and the rate of unidentified leakage allowed in the containment before the plant must be shut down.

Leak detection means which have been installed so far monitor the containment as a whole. Further, these detection systems have a stated sensitivity which is generally based on simple calculations. Little testing has been done to confirm the sensitivity of installed leak detection systems.

To implement leak-before-break properly, there must be an assurance that any crack which would lead to a break which would be of concern would be detected in time to avoid the break. This implies that leak detection systems installed must be able to detect a leak of appropriate size, and that the detection of such a leak would result in a definite signal being transmitted to the reactor operator that he must shut the plant down.

Chapter 3

METHODS FOR LEAK-BEFORE-BREAK ANALYSIS

3.1. Introduction

The concept of leak-before-break is based on the expectation that the pipes in a nuclear nuclear power plant are so tough that they would leak substantially before they would completely break. To demonstrate this characteristic, one postulates a flaw in the pipe being considered. Because systems subject to water hammer, fatigue, and stress corrosion cracking are excluded from consideration in leak-before-break, the mechanism by which this flaw would be formed is unknown. The assumption is that any flaw would begin small, then grow slowly until it reached a "critical" size, as defined in section 2.3.4. If the flaw were to reach the critical size for either a tearing or a plastic instability, the pipe would presumably fail catastrophically.

For leak-before-break to apply, one must show that the assumed flaw, under the influence of normal operating loads, would leak a detectable amount long before it would grow to critical proportions. One set of models which can be used to demonstrate leak-before-break is presented here. Limit-load type analysis will be used for calculating critical crack sizes, linear elastic conditions will be assumed when calculating crack opening sizes, and a homogeneous equilibrium model will be used to predict leak rates. Although other models exist, this set was chosen for simplicity and because the models are accepted as being reasonably accurate.

3.2. Mechanical Models For Cracked Pipes

Methods for the stress and fatigue analyses required in nuclear plant piping are generally accepted, and are set forth in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The only major change which leak-before-break would require in these analyses would be that all piping to be considered for leak-before-break must have some fatigue analysis. Whether the analysis is of the type depicted in the code for Class I systems, or some simpler version, the requirement exists, and is unlikely to change. The methods to be used are well understood.

Given the information from the stress analysis and a set of points to which leak-before-break is to be applied, the first step unique to leak-before-break would be to find a critical crack size for each location under consideration. Critical crack size depends on the maximum loading condition at the point, and on the mode of instability which is expected. The maximum loading conditions should be chosen as the worst conditions for which one would be concerned with leak-before-break. The authors of NUREG 1061, vol. 3, suggest that this condition would be normal operating plus Safe Shutdown Earthquake (SSE) loads. Choosing the mode of instability which would be expected may be difficult.

Tearing instability, which will be discussed in the section about fracture mechanics, is expected in materials which are not very tough, while the plastic instability is expected in the case of very tough materials. While one cannot aways predict which stability mode would dominate, some experts have suggested guidelines for simplicity. In Reference 18, the Piping Review Committee "recommends that a toughness comparable to or better than that of A106 Grade B carbon steel be demonstrated to justify using the limit-load approach." This criterion would make limit-load analysis applicable to all of the high energy systems in the subject plant for this study. Dr. G. Holman of Lawrence Livermore Laboratory has suggested that limit-load be used for austenitic steels, while tearing instability would be assumed to control in ferritic materials (Ref. 16). This second criterion would limit the use of limit-load analysis more than the first, since A106 carbon steel is a ferritic material.

In the present study, the only ferritic material to be considered is A106 grade B steel. Since the use of limit load analysis for this material is considered to be of marginal validity, the error introduced by using limit-load in this case is expected to be small. Reference 18 shows a maximum error of approximately ten percent on limiting load using this method on A106 steel. It is expected that uncertainties in leak detection capabilities and other analytical models would far outweigh the expected error introduced by using limit load analysis in this case. Further, calculations for tearing instability prediction require significantly more effort. Because of these factors, this paper does not present any results which rely on the use of tearing instability analysis.

3.3. Limit-Load Analysis

3.3.1. Critical Crack Size

Limit-load analysis assumes that the crack in a pipe does not propagate and cause failure. The strength of the specimen is then simply the strength of the remaining, uncracked cross section. Deformation and failure, then may be predicted by using traditional elastic and plastic relations applied to the particular geometry of the cracked pipe.

Plastic instability refers to a condition where the entire remaining pipe cross section (net crack) is assumed to be carrying a stress equal to the material flow stress, Sf, which is normally assumed to be the mean of the material's yield and ultimate strengths. From that assumption, references 18, 22 and 23 give the following characteristic equationλ

$$\frac{Pb \times \pi}{2 \times Sf} = 2\cos\left[\frac{\theta}{2} + \frac{Pm \times \pi}{2 \times Sf}\right] - \sin(\theta)$$

Here, Pm and Pb are the maximum stresses in the nominal case of the uncracked pipe, due to axial and bending loads respectively. θ is the angle which would subtend one half of the circumferential crack,. See figure 3.1.

Given the maximum loads to be considered, this formula determines, through iteration, the critical crack angle θ. The corresponding crack length is 2 R θ, where R is the mean radius of the pipe.

3.3.2. Crack Opening Area

Once the critical crack size has been determined, it is possible to apply a margin of safety on that size, then find the opening area of the reduced crack. This is necessary in order to calculate a leak rate. The crack opening area depends not only on the crack size, but also on the applied stresses, which deform the pipe. Most leak detection means require an ongoing leak, as opposed to brief occurences. Therefore, stresses associated with normal operating conditions should be used for this calculation.

In the recommended guidelines for leak-before-break set forth in Reference 18 a safety factor of two is applied to the critical crack size. Thus, the crack length found in the previous section must be divided by two before the appropriate crack opening area may be calculated.

Using the method from Reference 23, the crack opening area may be represented as the sum of the areas caused by axial and bending loads. Assuming most of the energy of deformation stored in the pipe is associated with elastic deformation, then for axial loads,

$$A(\text{axial}) = (Pm/E) \times 2 \pi R t G(\lambda)$$

where $G(\lambda) = \lambda^2 + .16\lambda^4$ if $0<\lambda<1$

or

$$.02 + .81\lambda^2 + .30\lambda^3 + .03\lambda^4 \text{ if } 1<\lambda<5$$

is the crack size parameter, and

$$\lambda = a (R t)^{-1/2} .$$

For deformation due to applied moments, the calculation is more detailed. Again from reference 3,

$$A(\text{bending}) = Sb/E \times (\pi R^2) Ib(\theta),$$

Where $Ib(\theta)$ is a ninth order polynomial in θ, and is given in Reference 23. The total crack opening area is the sum of the two calculated areas. The assumption here is that the deformation is generally elastic. Reference 23 also provides a method for correcting this result by using an effective crack length to take into account yielding in the zone of the crack tip. Several calculations performed using cracks of 1/2 critical size showed that the difference in areas calculated would be on the order of one percent.

3.4. Predicting Leak Rates

3.4.1. Flashing Flow

Making an accurate prediction of the leak rate from a narrow pipe crack presents a true challenge. Typically, if a pipe carries hot (temperature above 212oF, or 100oC) subcooled or saturated water, the fluid will flash to steam as it flows through the crack. The resulting two-phase flow is difficult to model accurately. In addition, because of the degree to which water expands as it turns to steam, the flow is typically choked, or sonic. A result from both experimental and theoretical work, however, leads to great simplification in this calculation.

3.3.2. Simplified Homogeneous Equilibrium Model

For subcooled water, if the flow path is long enough, as measured by L/D, the fluid pressure as it exits the crack is very nearly the saturation pressure for the initial fluid temperature. The flow through the crack, then is predominantly one-phase. Different researchers have found minimum values of L/D which range from 1.5 to 25. In all cases, the L/D for

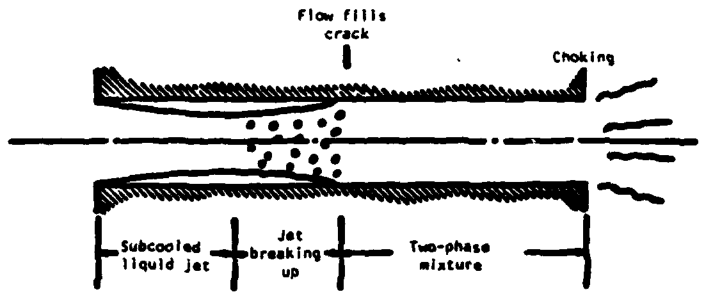

Figure 3.1 Two-Phase Flow Through a Long, Narrow Crack

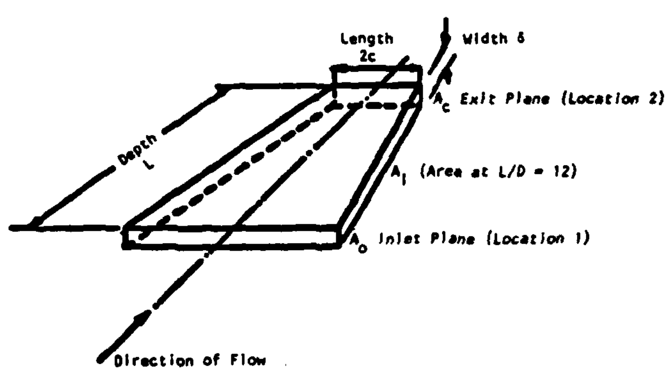

Figure 3.2 Geometry of a Convergent Crack

cracks under consideration here at least approach 25, and often far exceed that value. Figures 3.1 and 3.2 show the geometry of fluid flow through a crack, and the typical choke point. Reference 3 verifies the validity of the one-phase flow assumption, and recommends classical one-phase flow calculations for this case. The corresponding method is called the Simplified Homogeneous Equilibrium Method, or SHEM. Figure 3.3 demonstrates the accuracy of SHEM, which is comparable to the state-of-the-art computer code LEAK 01 for applicable cases (Ref 22). For incompressible flow, assuming pressure losses due to acceleration, entrance into the crack, and frictionλ

$$P(int) - P(exit) = G^2 \; v \; / \; 2 \; x \; (\; 1 + 1/C^2 + fl/D \;)$$

where the orifice coefficient C=.61 (ideal) is typically assumed. The friction factor, f, is given by the modified Von Karman relationλ

$$f = (\; 2\log(D/2K) + 1.74 \;)$$

Here, l is the flow path length through the crack, or the pipe wall thickness. The wetted perimeter is

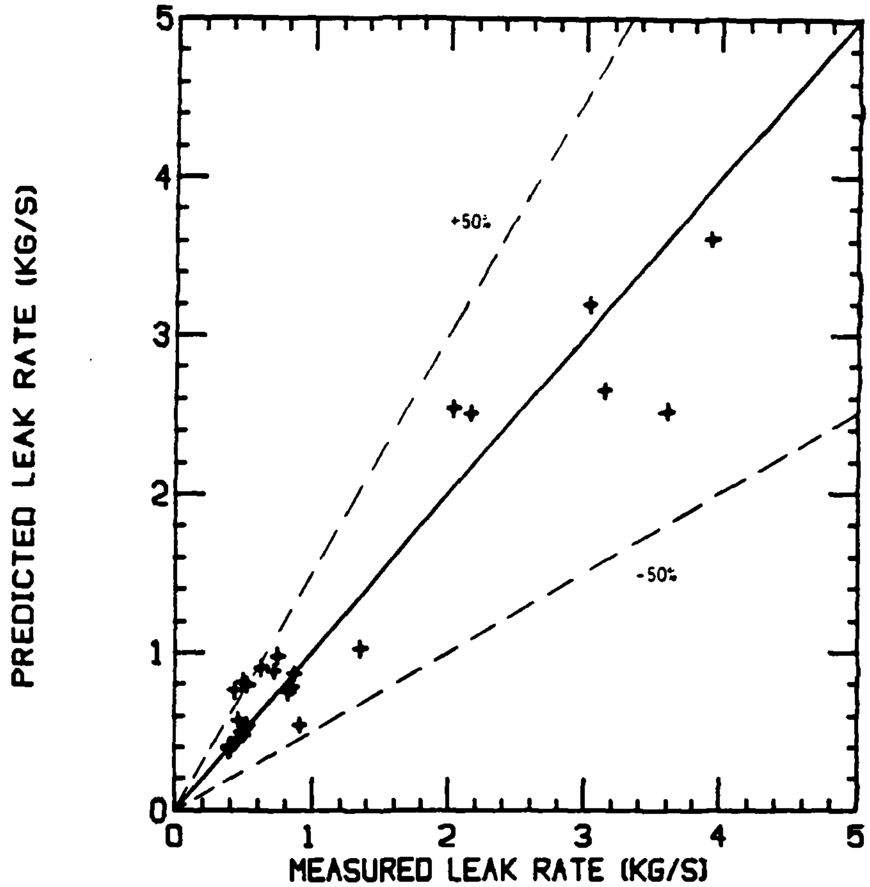

Figure 3.3
Comparison of Mass Flow Rate Predictions Using Simplified Homogeneous Equilibrium Model with Measured Flow Rates Through Simulated Cracks

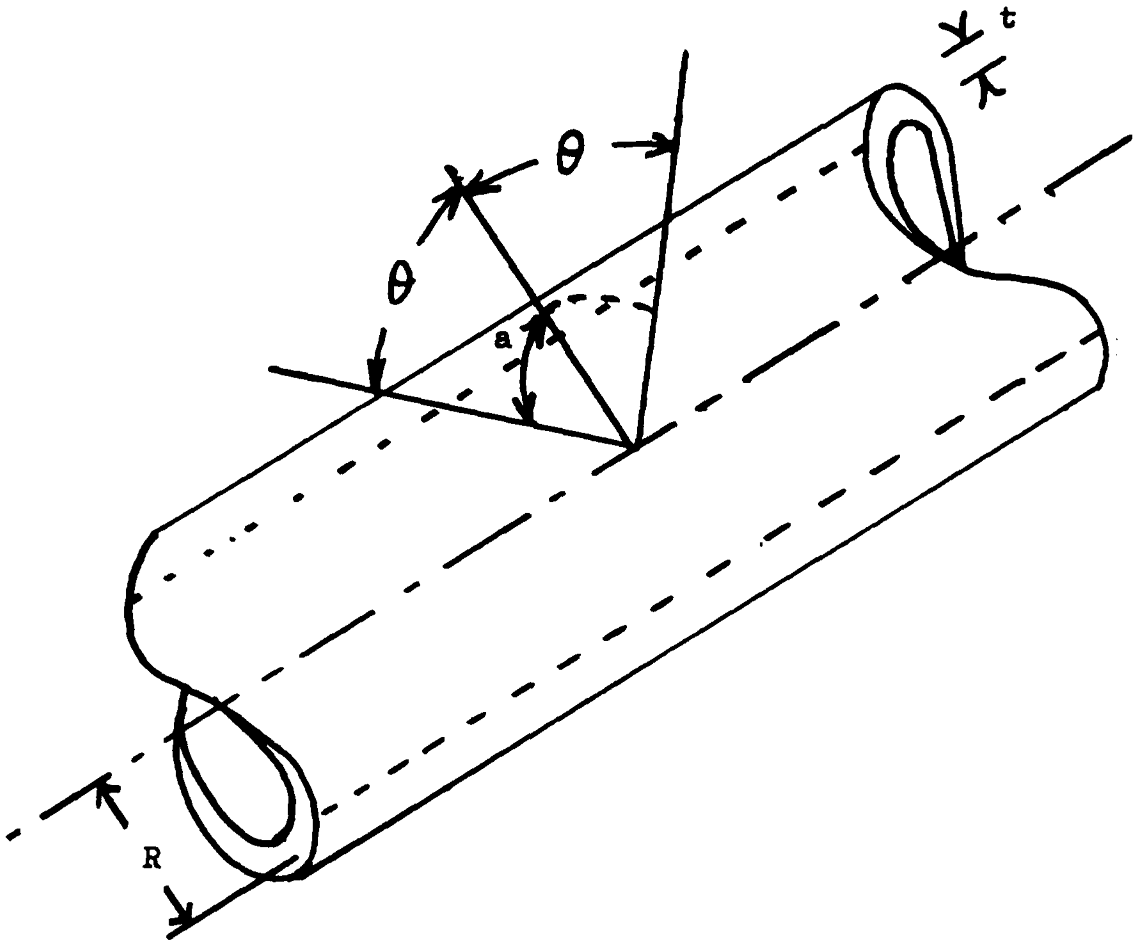

Figure 3.4: Geometry of a Cracked Pipe

approximately 2 (COD + L). Since the hydraulic diameter is given by

$$D = (4 \times \text{Area} / \text{Wetted Perimeter})$$

then hydraulic diameter in this case is given approximately by

$$D = \frac{(4\ \text{COD}\ L)}{2\ (\text{COD} + L)}$$

$$= 2 \times \text{Area} / L\ , \qquad \text{COD} \ll L$$

(See Figure 3.4)

Reference 3 suggests a value of K=0.0051mm (=2E-4 in) for stress corrosion cracks.

If the fluid in the pipe is not subcooled, then two possibilities exist for leak rate prediction. Computer code LEAK 01 was developed for EPRI, and predicts two-phase flow rates through cracks. As a simpler alternative, upper and lower limits can be set on the actual leak rate. By assuming simple, nonflashing one-phase subcooled flow, one can calculate an upper bound. Using an equation for saturated steam flow through an orifice, such as the Grashof expression,

$$G = .0165\ p^{.97}\ \text{lbm/sec-in}^2,$$

with p in psig, a lower bound for flow rate may be found. This specifies the leak rate to within about an order of magnitude, which may or may not be sufficient. If the lower bound would be detectable, leak-before-break would apply. If the upper bound would not be detectable, leak-before-break would not apply. If the detection threshold lay somewhere between the upper and lower bounds, then a two-phase caluclation would be required.

Having applied a safety factor on crack size, the resulting leak rate is calculated. Reference 18 suggests that a margin of 10 be applied to this leak rate for conservatism. When the suggested margins are applied to crack size and leak rate, the calculated leak rate is typically conservative by two orders of magnitude, when compared to the leak rate calculated directly from the critical sized crack. Figure 3.5, taken from reference 22, demonstrates the effect of this conservatism. In fact it is significant that the leak rate incrases very rapidly as crack size approaches the critical size. This is because the deformation increases substantially as the crack

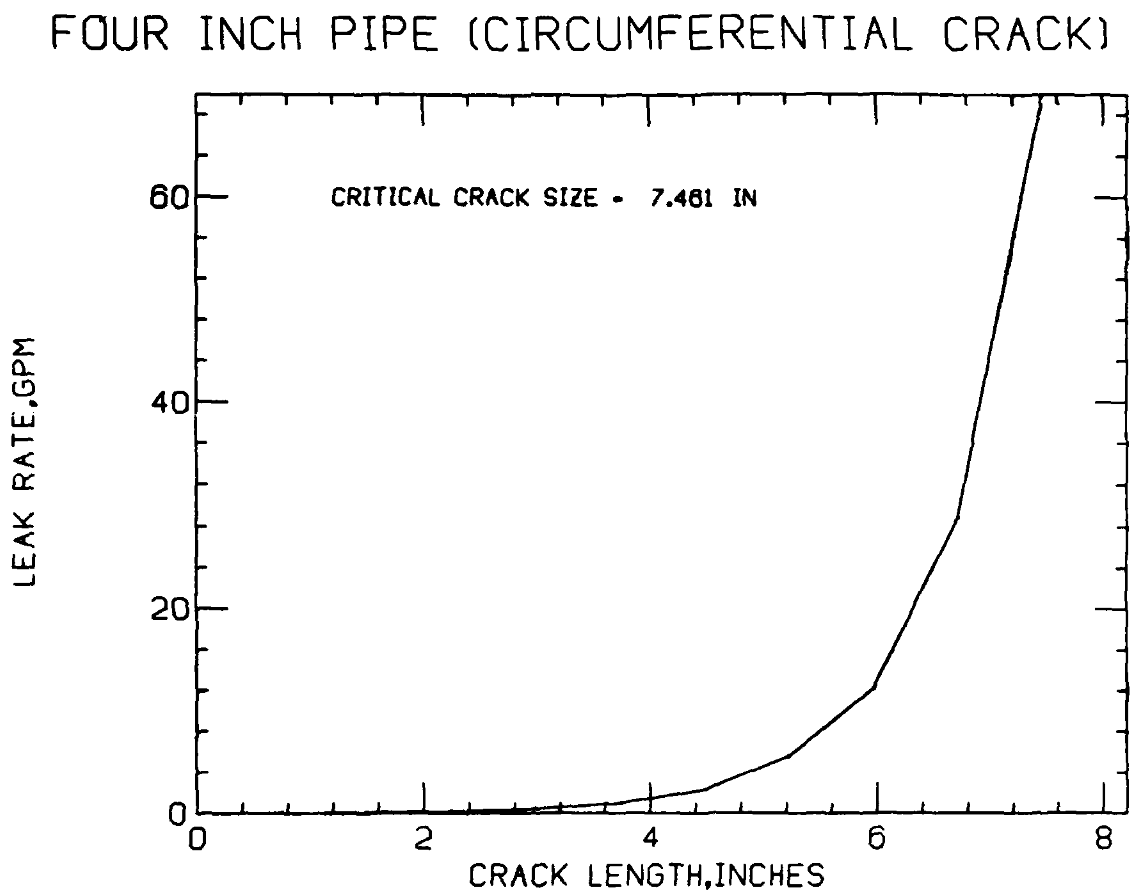

Figure 3.5. **Effect of Crack Size on Leak Rate**

approaches critical size, regardless of what the critical size really is. In other words, even if the critical size were predicted incorrectly, the leak rate would become large when the actual critical size is approached. The Nuclear Regulatory Commission is presently studying the uncertainties of the various models involved, to determine whether these margins should be reduced, and by how much.

3.4. Fracture Mechanics

3.4.1. Fracture Mechanics

To ensure that a crack is stable, it is necessary to consider both tearing and plastic instabilities, as defined in section 2.3.4. As mentioned previously, the plastic instability is the expected failure mode for tough reactor piping. To demonstrate leak-before-break using limit-load calculations, however, one must argue that this analysis is justified. A basic understanding of fracture mechanics is required.

In any load-bearing member, such as a pipe, cracks and other geometric nonuniformities lead to regions of high stress, or stress concentrations. Around these stress concentrations, the member may fail locally without even approaching the limiting strength of the overall member. This forms the basis for crack growth, or fracture mechanics.

3.4.2. J-Integral

When a structural member such as a pipe is mechanically stressed, it stores potential energy. This is analogous to compressing a spring. In the region of a crack, the distribution of stresses may be complex. Nevertheless, a convenient measure of the volume-averaged potential energy stored in the stressed region is the "J-Integral". Theoretically,

$$J = \int_{\Gamma} (Wn_1 - S_{ij}n_j u_{i1})$$

where the path of integration is any path which encloses the tip of the crack, W is the stress-strain energy density, S is the stress, n is the outward normal unit vector, and u is the displacement. This expression is rarely used in this form, however. If the majority of the stressed region is deformed elastically, the energy density is simply stress x strain. Many people are familiar with the stress intensity, K ($K = Snom(\pi a)^{1/2}$ times some geometric factor, where a is the crack length). When the deformation is small and most of the energy is stored as elastic deformation, then $J = K^2/E$, as one might expect.

In cases where plastic deformation is important, the plastic contribution to J can be calculated as a function of material properties, geometry, and the magnitude and configuration of the applied load. The effects of geometry and load configuration have been expressed in the form of several non-analytic functions, some of which are given in References 20 and 23, for some cases of cracked pipes. The J integral is used both in predicting stable crack growth rates, and in calculating the conditions which would lead to unstable crack growth.

The lowest value of J which results in crack extension is called J_{1C} for the material. If, through analysis, it is found that J is greater than J_{1C}, it becomes necessary to determine how fast the crack would grow. The appropriate fatigue crack growth analysis is given in Section XI of the ASME Boiler and Pressure Vessel Code, and is already required for all Class I piping systems in the plant. Class I systems are those which form the pressure boundary for the primary coolant. If crack growth is predicted to be very small during the period between in-service inspections, then it is not of concern for leak-before-break, and one need only satisfy the criterion of crack stability.

Just as the force exerted by a spring is the derivative of its potential energy, $dU/dx = k|x-x_0|$, a measure of the driving force for crack extension is the derivative of the potential energy density dJ/da. If this driving force is greater than the material's inherent capaacity, the crack will grow. If dJ/da does not decrease with crack size, the crack will grow unstably. In this case, the larger the crack grows, the lower the energy state. The stable state, then, is that of a broken pipe.

3.4.3. Tearing Modulus

The standard test for the tearing instability is to compare the value of the applied "tearing modulus", Tapplied, to the measured value of the material "tearing modulus", Tm.

$$T = (E/Sf^2) \times (dJ/da)$$

Again, one must use complicated geometric functions to find values for J. Typically, either J or T is plotted graphically as a function of crack length or applied load. The curves are compared to experimental curves corresponding to the material being analyzed to

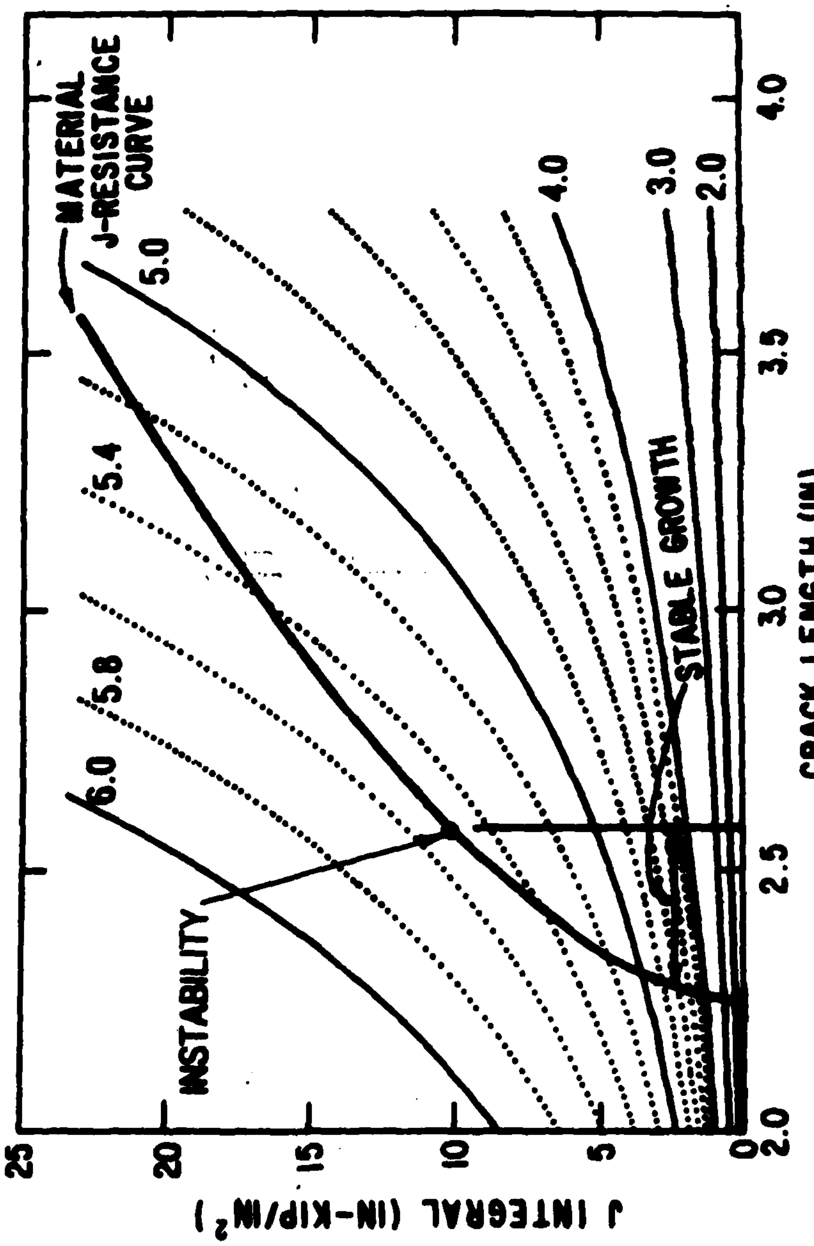

Figure 3.6. Demonstration of the Use of J-Integral to Predict Tearing Instability

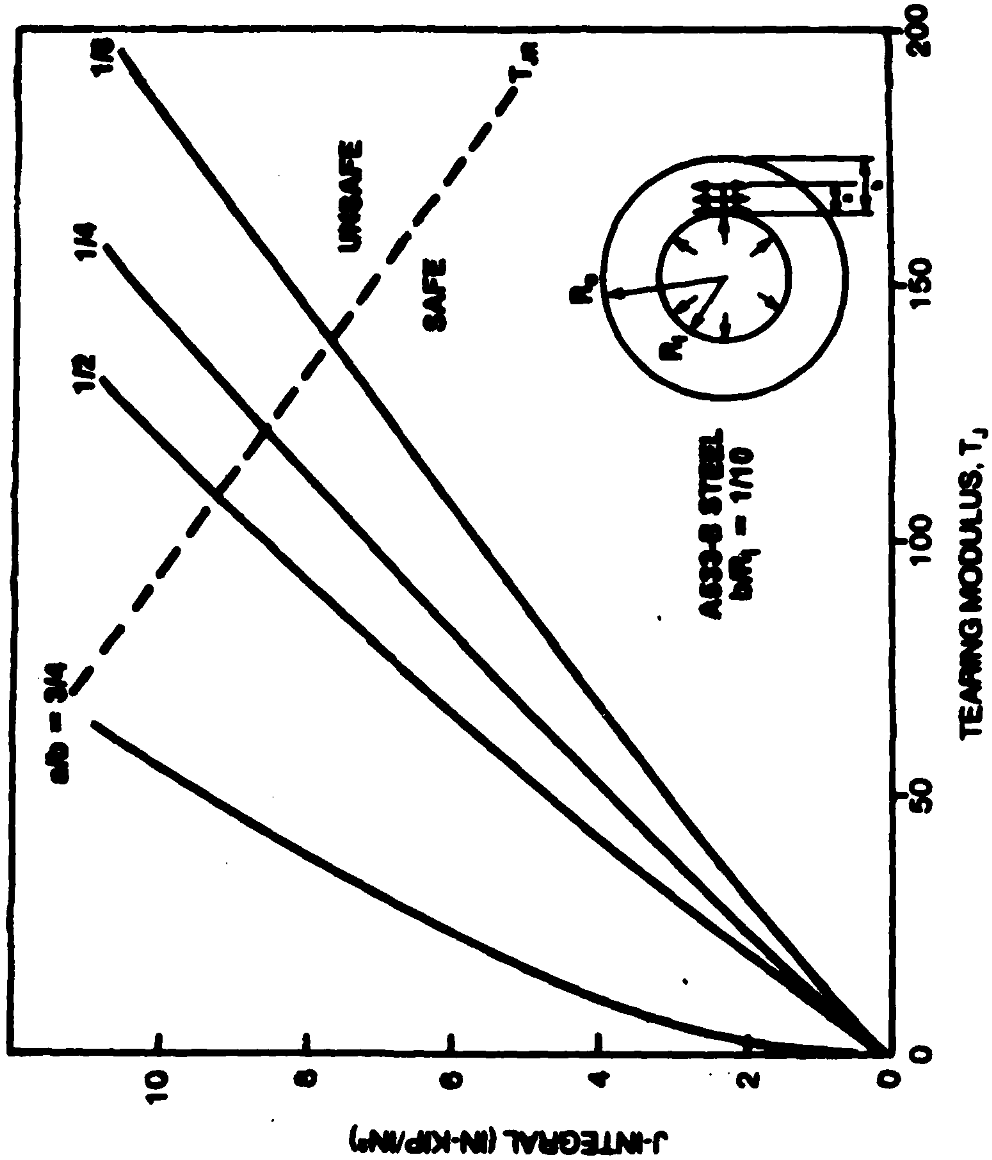

Figure 3.7. Demonstration of the Use of Tearing Modulus to Predict Tearing Instability

find the load or crack size which leads to a tearing instability. Figures 3.5 and 3.6 are sample results from such an analysis. They demonstrate how to find the points of instability. In Figure 3.5, the slopes of the characteristic J curves for the loading and for the material are compared. If the slope of J vs crack length for the material is lower than the same value for applied load, then the condition is unstable. In figure 3.6, the values of T are compared and, again, if the applied T is too high, an unstable condition results. The analysis is difficult and costly. Testing costs for just one material are an estimated $14,000 (Ref 34).

Simple yet accurate methods exist for demonstrating leak-before-break in a reactor pipe system. If one can justify using limit-load analysis for crack sizes, and one-phase flow calculations for leak rates, then the only significant additional effort required in leak-before-break justification is a "Class I" fatigue analysis. Where the validity of these assumptions is questionable, one might either use large safety margins, or perform the more detailed analysis. In most cases, even using fracture mechanics and two-phase flow calculations are worthwhile, in order to eliminate consideration of pipe rupture in nuclear plant design.

CHAPTER 4

APPLICATION OF METHODS

4.1. Introduction

The simplified methods for leak-before-break have been applied to the systems of the example PWR nuclear power plant in order to assess the potential impact of leak-before-break on plant design. Four specific analyses were performed. First, all piping systems in the plant which have pipe rupture protection installed were examined, and a representative leak rate was calculated. Next, this result was used to relate nominal pipe size to expected leak rate. In order to motivate the definition of a worst-case condition for leak-before-break, a specific pipe was chosen, and its maximum stress level was varied. The predicted leak rate from a crack of one half critical size was correlated to the maximum applied stress. Finally, the given methods were applied to a specific portion of the plant, and the potential for leak-before-break application in that specific location was examined in detail.

4.2. Survey of High Energy Systems

4.2.1. Assumptions

All high energy systems of nominal pipe size greater than 1" (2.55 cm) must be evaluated for hazards which they might pose to essential safety systems. A high energy safety system is one which normally carries fluid with a temperature greater than 200oF or 93C, or a pressure greater than 275 psig, or 1.96 Mpa. Essential safety systems are those systems which would be required to operate to ensure that the plant could be shutdown safely.

Table 4-1 summarizes the high energy systems in the example nuclear plant which require pipe restraints, according to Chapter 3.6 of the Final Safety Analysis Report (FSAR), Reference 11.

Information about operating conditions and material properties was taken from the actual piping system specifecations, reference 29. The loading conditions on the pipes are location specific, and depend on the particular state of the plant. A standardized loading configuration was assumed in order to calculate expected leak rates from the different systems. Several references indicated that the major contribution to pipe stress is bending moment. Further, from results cited in Reference 22, circumferential cracks were determined to be the

Table 4.1

SUMMARY OF RESULTS FROM SIMPLIFIED
LIMIT LOAD ANALYSIS: EXAMPLE PLANT.

SYSTEM NAME	NOM. PIPE SIZE (IN)	NO OF RESTR	LK RATE CS 1(GPM)	LK RATE CS 2(GPM)
Primary	31	8	-	-
Coolant	29	4	-	-
	27.5	4	29.6	247
	8	40	1.10	17.5
Main Steam	30	48	11.9 - 45.7	52.3-261
Feedwater	20	40	6.57	82.7
	18	4	5.188	66.7
	8	20	.852	14.63
Pressurizer	14	7	1.94	30.4
	6	2	.407	8.95
Res Ht Rem	12	4	1.14	16.4
and LPSI	10	4	1.05	11.9
	6	2	.226	4.18
Stm Gen BD	4	24	.0172	.409
Aux FW	4	12	.230	4.09
CVCS	3	11	.036	1.13

limiting case for leak-before-break.

Typically, the stress acting on a circumferential crack may be considered to be the sum of axial stress due to internal pressure, or primary membrane stress, and the stresses induced by bending moments caused by deadweight and reaction forces. For this example, the axial stress was assumed to be equal to the primary membrane stress, while the total stress was taken to be equal to 3/2 Sm, where Sm is an allowable stress which is defined by the ASME Code for each given material and temperature. This quantity was chosen because it corresponds to the maximum allowable value for total primary stress. Primary stress does not take local (geometric) effects nor displacement controlled stresses into consideration. This example may not correspond to the worst case situation with respect to leak-before-break, nor does it represent an actual point. It does, however, provide a standard for comparing systems, and is expected to approximate the actual stress levels in each system.

4.2.2. Method

The computer program "Break Free" was written which incorporated the limit-load equation for plastic instability, an elastic crack opening area calculation from Reference 23, and the Simplified Homogeneous Equilibrium Model, all discussed previously. The program is given as Appendix B. Given the stresses from either a stress analysis or from the above assumptions, the code predicts the critical crack size, divides it by two, then calculates the crack opening area. The latter calculation is based on the assumption that deformation is primarily elastic. Reference 23 gives a method for correcting for the effect of plastic deformation, but it was found to give less than a percent difference in calculated crack opening area. The program then calculates hydraulic characteristics of the crack and, given fluid properties, calculates leak rates. The leak rates are expressed in gallons per minute, based on a standard 8.33 pounds-mass per gallon

Calculations for Table 4.1 were performed under two assumptions. In the first case, the stress levels mentioned above were applied when considering critical crack size, but only the axial stress due to internal pressure was assumed when calculating crack opening area. This assumption was taken for conservatism, since the internal pressure could be expected to act continuously during operation, while some other loads might be periodic thus not always contributing to the leak rate. The second case assumed that the maximum bending stress was also the operating stress. This demonstrated the best case possible in terms of leak-before-break, and provided an upper bound on leak rates to be expected from a leak-before-break analysis of the given system.

During this investigation, the various sizes of pipes were broken down into three groups. Large pipes, which includes pipes 18 inches (46 cm) in diameter or larger, are mainly the coolant, steam and feedwater pipes. Medium sized pipes, including pipes from 8 to sixteen inches (21-41 cm) in diameter, are large auxiliary lines, such as pressurizer surge and coolant bypass lines. Finally, the small lines, including those from 2 to six inches (5.1-15 cm) in diameter, are utility lines such as steam generator blowdown, auxiliary feedwater, pressurizer spray, and chemical and volume control lines.

4.2.3. Results

Table 4.1 and Figure 4.1 reflect a summary of leak rates calculated in the systems survey. The leak

TABLE 4.2

DETECTION CAPABILITY REQUIRED

FOR LEAK-BEFORE-BREAK QUALIFICATION

-CASE ONE-

PIPE SIZE (IN)	LEAK RATE (GPM)	DETECTION SENSITIVITY REQD (GPM)
LARGE 18-33 IN	5 - 45	.5 - 4.5
MEDIUM 8-16 IN	0.85 - 2.0	0.085 - 0.2
SMALL 2-6 IN	0.02 - 0.4	0.002 - 0.04

-CASE TWO-

PIPE SIZE (IN)	LEAK RATE (GPM)	DETECTION SENSITIVITY REQD (GPM)
LARGE 18-33 IN	50 - 250	5 - 25
MEDIUM 8-16 IN	12 - 30	1.2 - 3
SMALL 2-6 IN	0.4 - 9	0.04 - 0.9

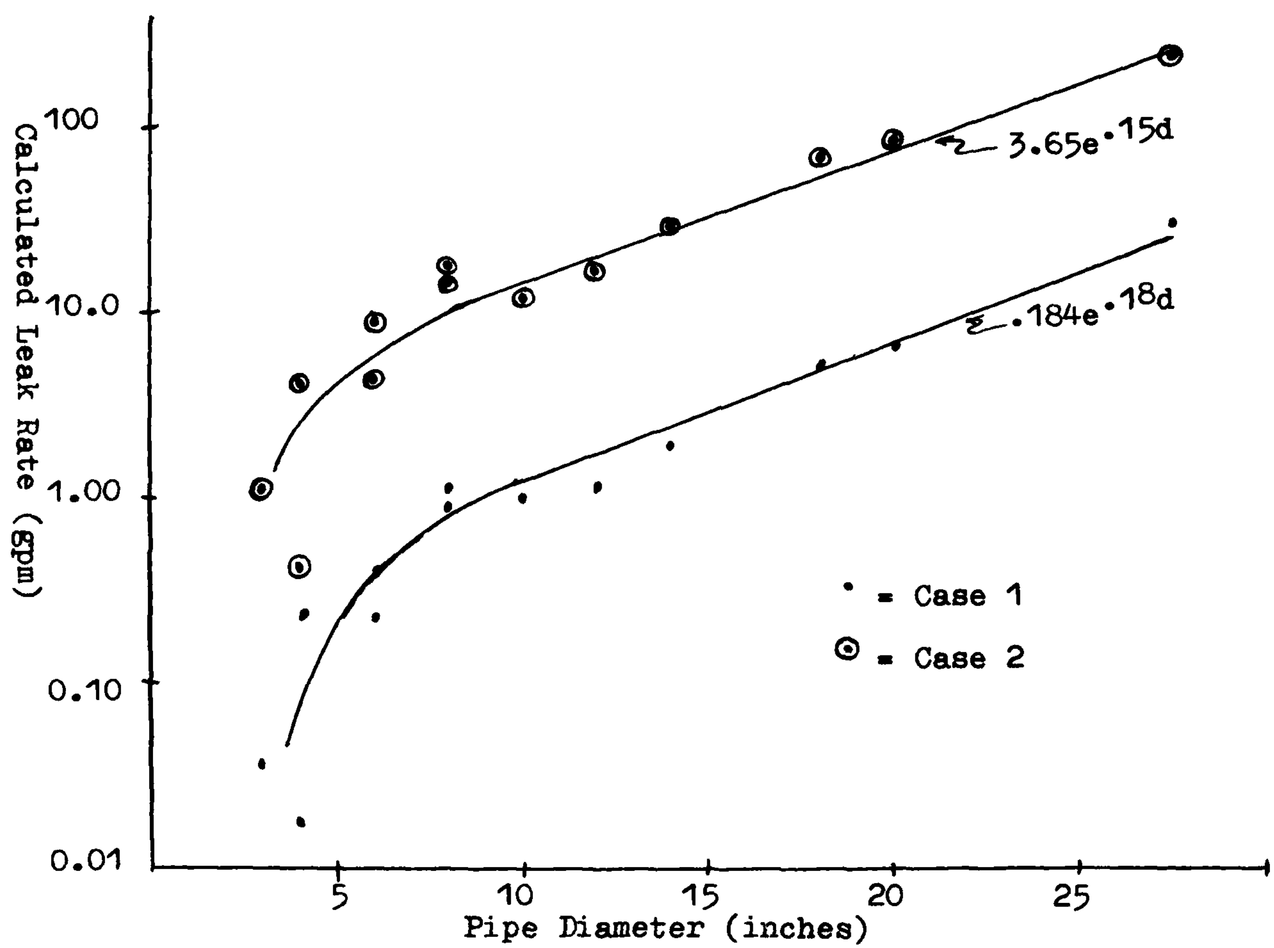

Figure 4.1. Leak Rate versus Pipe Size

rates vary significantly, reflecting the different assumptions of cases one and two. In all cases, the maximum stresses determined critical crack size. "Case 1" assumed primary membrane stress alone would cause crack opening, while "case 2" assumes that all operating stresses from the ASME stress analysis contribute to crack opening, and therefore to leak rate. The leak detection sensitivity requirements shown on Table 4.2 represent a factor of safety of ten applied to leak rate, as suggested in reference 18.

4.2.4. Conclusions

Leak detection systems currently installed in light water reactors have stated sensitivities on the order of 1 gallon (3.8 liters) per minute detected within between one and four hours. Combining this result with table 4.2 results, it is apparent that systems of large pipes should have little trouble in implementing leak-before-break criteria, while systems of medium pipes would depend on the acceptance of case two assumptions, or improved leak detection capability. The key assumptions in this analysis are the relationship of maximum and steady state stresses (case 1 or case 2 assumptions), and the factor of safety applied to leak rate. In the most conservative case, only large systems may meet leak-before-break criteria, while under more realistic assumptions, medium-sized systems could meet the criteria without requiring modifications to present leak detection systems.

4.3. Effect of Stress on Leak Rate

4.3.1. Assumptions

If leak-before-break could be justified by examining one sample case for each high energy piping system, it would result in a great savings in effort, and therefore money. Such a situation would require that a worst case could be defined with respect to leak-before-break, which would correspond to the highest probability for undetected crack growth.

As one step toward defining a worst case, an analysis was performed on a specific pipe system. The applied bending stress was varied, and the expected leak-before-break leak rate was calculated. If high maximum stress should correspond to low leak rate, then the highest stressed locations would be the worst cases, since the probability of a crack occuring would be high, while the probability of detecting it through leak detection would be low. If, however, some lower state of maximum stress resulted in a critical crack with lower corresponding leak rate, then the worst case would be more difficult to define.

4.3.2. Method

The six inch (15 cm) pressurizer spray line was chosen as the subject for this analysis. The maximum bending stress was varied from zero to 30 ksi (207 Mpa). This upper stress level corresponds to slightly greater than 3/2 Sm. Figure 4.2 demonstrates the dependence of predicted leak rate on maximum stress. The two curves correspond to the cases one and two, where bending stresses are included and not included, respectively, for leak rate calculation. Note that higher stress invariably corresponds to a larger critical crack. As expected, if only the membrane stress is available to cause crack opening, then the smaller the crack, the smaller the leak rate will be. If the bending stress is included, however, the leak rate is highest when bending stress Pb = Sm = 18.1 ksi (124 Mpa). Thus, if the highest stresses encountered in a system approach the limit, then a worst case would correspond to the point of highest stress. If, however, the highest stresses locations have a maximum stress which is near or below Sm, then the places where cracks are most likely would not correspond to the lowest leak rate from a critical crack.

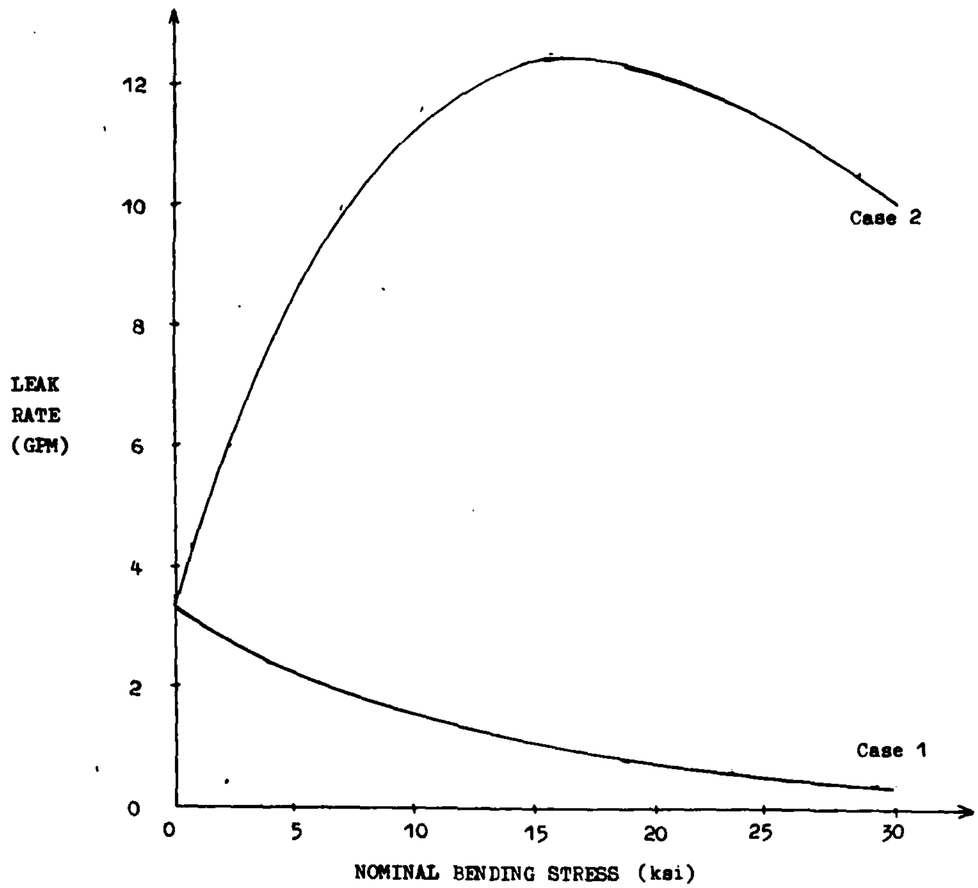

Figure 4.2. Predicted Leak Rate Versus Maximum Applied Stress

4.4. Analysis of Actual Area

As a demonstration of the potential effects of leak-before-break on an existing light water plant, the analysis was applied to a specific location within the example plant. Specifically, the lower portion of the cubicle which houses the steam generator in coolant loop B was chosen because of several features. The area of consideration is somewhat enclosed by the floor, the cubicle walls, and the bottoms of the steam generator and the reactor coolant pump. Pipes from several high energy systems run through the area, resulting in a high density pipe restraints. Finally, this particular cubicle contains a restraint on the pressurizer surge line, which does not enter the other steam generator cubicles.

Sixteen pipe whip restraints of various design are housed in the area to mitigate the predicted effects of eleven postulated pipe breaks. Table 4.3 summarizes the predicted leak rates from cracks of one-half critical size, using the actual calculated stress data from the ASME stress analysis. Again, the leak rates are predicted for both case 1 and case 2 assumptions. In all actual cases, the maximum bending

Table 4.3
LEAK RATES AT POSTULATED
BREAK LOCATIONS: LOWER
STM GEN CUBICLE B

LINE NAME	SIZE (IN)	BREAK NUMBER	CRIT CRACK SIZE (IN)	LEAK RATE PRESS. ONLY	LEAK RATE OPER STRESS
Surge	14		19.126	1.715	7.406
Bypass	8	14	11.241	1.675	11.345
	8	15	12.132	3.717	15.706
	8	16	11.050	2.737	16.985
	8	17	11.863	3.453	15.941
LPSI	10	2	16.571	11.010	19.66
(Low	10	3	13.819	6.152	30.999
Pres	10	4	13.927	6.307	31.153
Inj)	10	5	16.625	11.126	19.542
Stm Gen	2		2.756	7.068E-4	.0683
Blwdn	4		7.211	.0870	.2820

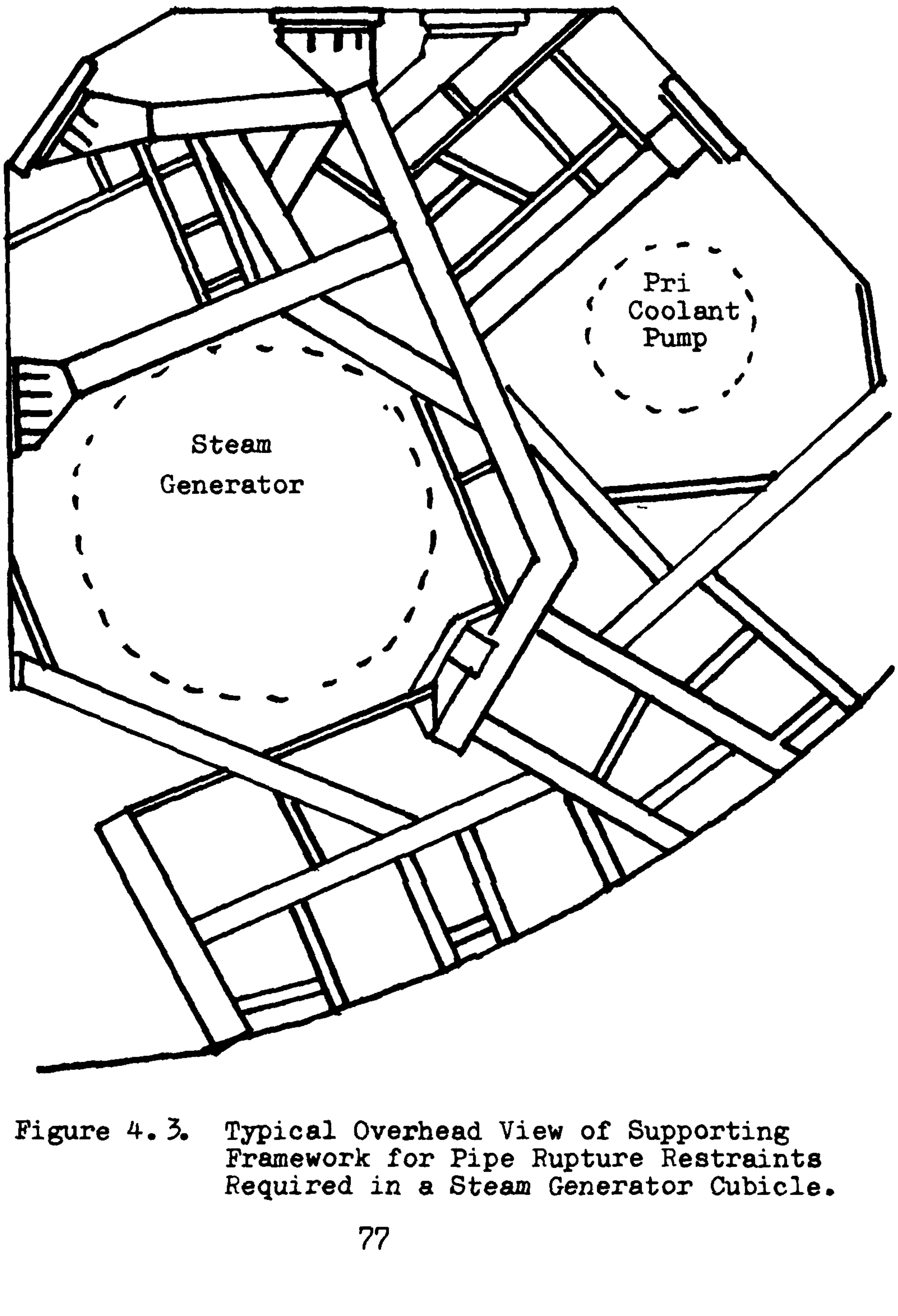

Figure 4.3. Typical Overhead View of Supporting Framework for Pipe Rupture Restraints Required in a Steam Generator Cubicle.

stress is much smaller than 3/2 Sm. The critical crack sizes are therefore larger relative to pipe diameter than in the calculations in section 4.2. Further, because the ASME operating stresses are lower than the maximum stresses used in section 4.2, there is less of a difference between case 1 and case 2 assumptions.

ASME calculated normal operating stress takes into account primary membrane stress, and stresses due to deadweight, thermal transients, and operating basis earthquakes. An examination of the actual stress data indicates that operating stress sometimes includes a significant contribution from transient effects, while in other cases, there is very little contribution. In the former case, the assumptions of case 1 would be proper, while in the latter case, the case 2 assumptions would be valid. Because stress analysis results usually express operating stresses as a combination of steady-state and transient effects, case 1 assumptions should be used in general for conservatism. If the threshold of leak detectability lies between the results of case 1 and case 2 calculations, the stress analysis calculations must be examined to determine the contribution due to transient effects. Only operating stresses which act over time periods long enough to affect leak detection should be included when calculating leak rate.

For a given pipe segment, the postulated break with the lowest leak rate determines whether leak-before-break would apply. If case 1 assumptions are used and the recommended safety margin of 10 is applied to leak rate, only the accumulator safety injection line would qualify, for a reduction of 2 restraints out of the 16 included in the area of consideration.

If one assumes the full calculated value of operating stress, however, 14 of the 16 restraints could be removed, leaving restraints only on the 4 inch (5.1 cm) steam generator blowdown line, These two restraints are relatively small.

A study of stress analysis calculations indicates that a significant portion of the operating stress on the pressurizer surge line is due to seismic loading, while there is little seismic contribution to stresses in the safety injection line. This suggests that the restraint on the surge line might not be removable, since its leak rate under case 1 assumptions is only 1.7 gallons (6.5 liters) per minute. Nevertheless, the restraint on the surge line requires little supporting structure. In future plants, the supporting framework shown in Figure 4.3 could be virtually eliminated.

CHAPTER 5

EFFECTS OF LEAK-BEFORE-BREAK ON PLANT DESIGN

5.1. Introduction

Piping systems in nuclear power plants are designed using the same basic principles as those in other applications. Because of the special problems of radiation and the potential consequences of a hardware failure in a nuclear facility, however, some additional requirements are placed on the design process. These additional requirements include fatigue analysis, the postulation of breaks, and an analysis of the potential effects of such breaks. Because of the requirement to postulate breaks, protective measures must be taken to ensure that such theoretical breaks would not degrade the plant's ability to be shut down safely.

Design considerations for pipe rupture affect not only pipe system design, but other plant features as well. The effects of pipe rupture have been divided into the two categories of dynamic effects and others. While protection measures against dynamic effects, such as pipe whip are the main concern of leak-before-break, the potential exists for extending the argument that breaks would not occur to other effects, such as compartment pressurization, in the future. To examine the potential for changes that leak-before-break might have on light water reactors, then, one must consider both the short-term effects of excluding consideration of dynamic effects, and the long-term possibility of excluding other effects.

Another difference between short-term and long-term effects is that nuclear plants which have already been built, already have the required protection features included. Plants which have been substantially begun have design work completed, and in many cases, have the hardware already installed. There is less to be gained by removing devices which have already been installed than by having no requirement to design or build them at all. Certain design features would not be removed. The containment building must be designed to contain the pressures resulting from a full double-ended guillotine break of the largest line. If that requirement were removed, then future containments could be smaller and less substantial. Once built, however, one would not expect a nuclear plant owner to tear down the old containment building to replace it with one of lesser strength. For these reasons, design considerations for dynamic effects

Table 5.1

PIPE RUPTURE EFFECTS AND PROTECTION MEASURES

1. Pipe Whip
   - Physical Separation
   - Barriers
   - Pipe Rupture Restraints

2. Fluid Jets
   - Physical Separation
   - Barriers
   - Jet Impingement Shields
   - Environmental Qualification of Instruments

3. Loss of Coolant
   - Safety Injection
   - Emergency Core Cooling System

4. Containment Pressurization
   - Containment Integrity

(Continued on next page)

Table 5.1 (continued)

5. Compartment Pressurization and Fluid Jet Thrust

   -Compartment Integrity

   -Compartment Ventilation

   -Component Support Integrity

6. Assymetric Blowdown

   -Would require complete re-design of internal portion of reactor pressure vessel. Need not be considered, however, because cracks propagate slowly enough will be treated separately from others, and the effects of leak-before-break on future plants will be separated from the effects on present plants. Table 5.1 summarizes the effects of pipe rupture and the design requirements to protect against those effects.

5.2. Design for Dynamic Effects of Pipe Rupture

5.2.1. Protective Measures

The dynamic effects of pipe rupture include pipe whip, the generation of missiles, and fluid jet impingement. To protect against these effects, several measures may be taken, all of which are associated with the pipes themselves. These measures include physical separation, separating barriers, pipe rupture restraints, and jet impingement shields. Before considering how these measures affect nuclear plant design, it is important to consider first the methods which are used in designing pipe systems.

5.2.2. Pipe System Design

When selecting a pathway for pipes to follow, certain considerations are important, even for high energy nuclear piping. First, pipes must begin and end at the components which they connect, e.g. steam generator or coolant pump. From the serviced component, pipes are routed as directly as possible to the nearest substantial structure. Designers place pipes along solid structure in order to leave the interior building space less cluttered, to allow for pipe support, and to provide a reference for organizing similar pipes into groups or gangs (Ref 14).

The next consideration is the protection of safety systems from the effects of pipe rupture. Pipe whip and jet impingement are the two important hazards. Any system in the plant which is required for safe reactor shutdown must be protected against these two effects in the event of a pipe break. Three methods may be used to provide this protection: separation, barriers, and restraints. Once these have been applied to the required degree, the piping layout may be "fine-tuned" to eliminate any interferences among pipes, and between pipes and other components.

5.2.3. Physical Separation

The most effective protection measure is to separate the high energy piping systems and essential safety systems with enough distance so that a break in the high energy line would not affect the capability of the safety system to perform its mission. Also, the chances for failure of the safety and high energy systems due to a common cause are decreased, and the probability of damaging one system in the course of building, inspecting or maintaining the other system are minimized. The disadvantage is that applying the principle of separation requires space, and can make piping runs longer. In a typical Pressurized Water Reactor plant, separation may add an estimated ten to fifteen percent to the length of applicable piping systems (Ref. 14). Certainly each foot of added piping is as likely to fail as the original pipe, except that added bends may contribute to the system's flexibility.

5.2.4. Barriers

The second method for providing protection against pipe rupture effects is to impose barriers. Where it is not practical to re-route high energy piping enough to avoid safety systems completely, the two may be separated by a barrier, such as a wall. Examples of this concept include the concrete walls between steam generator "cubicles", and small concrete walls inserted between a pipe and some vulnerable system. This protection measure explicitly entails some hardware costs, but can be less costly than moving large groups of pipes or safety equipment. Another disadvantage to using this measure is that adding structure in the plant limits the movement of personnel and equipment in the vicinity of that structure. This makes construction, maintenance and inspection tasks more difficult Each of these activities, therefore, becomes more costly and less effective. An advantage of using barriers, however, is that one wall may protect a group of targets from a group of potential pipes without requiring any relocation of components. Typically, only one or two small walls are added to a plant design specifically as barriers for pipe rupture protection (Ref. 14), so the overall effect of barriers on reactor design is minor.

5.2.5. Pipe Rupture Restraints.

As a last resort, strong structures may be added at specific pipe break locations. These structures, known as pipe rupture restraints, are placed so that the pipes could not impact on the essential safety system, regardless of where the pipe might break. Rupture restraints might be rigid supports which limit movement of the pipe to very small displacements, or they might be energy absorbing devices, such as ductile straps or crushable bumpers, which would limit the movement of the pipe by absorbing kinetic energy. In either case, the restraints are designed to avoid restraining the thermal expansion of the pipes. For this reason, rigid restraints are generally "snubbers", or dashpot type mechanisms which resist sudden motion but not gradual movement. Energy absorbing restraints are generally installed with some small gap between the pipe and the restraint.

Pipe rupture restraints must be capable of withstanding tremendous reaction forces from the thrust of the fluid which would escape from a large pipe break. Where restraints are located near a wall or floor, special anchorages must be installed to provide the required strength. If restraints are required in a location which is not near a wall, a substantial steel framework must be constructed which is strong enough to support the full anticipated load on the pipe restraint. The legs of such a space frame must also be anchored to the concrete floor or wall. The framework and anchorages required for pipe rupture restraint intallation make a significant contribution to plant design and construction effort, and to the material requirements imposed by pipe rupture consideration. The steel space frames substantially limit access to anything in the vicinity.

As mentioned above, any pipe rupture restraints must be installed such that they would not inhibit thermal expansion of the pipes. It is not always possible, however, to predict precisely every state that the plant will be in during operation. If some error is made in the prediction of operating states, or if the restraints are installed incorrectly, the pipe restraints may restrict the thermal expansion of the pipes, resulting in potentially high stresses. While restraints are installed to protect the plant in the unlikely event of a pipe rupture, they may, in fact, increase the chances for pipe cracks or rupture by increasing the level of applied stresses.

The one advantage of pipe restraints is that they are the simplest pipe protection measure to remove, should they be no longer required. In some cases, the supporting structure, or even the restraint itself may be too massive to remove completely, but by removing a portion closest to the pipe, it is possible to allow improved access to the pipe for inspection. This would improve plant reliability and reduce the radiation exposure to workers who perform inspection or maintenance tasks on the pipe.

5.3. Design for Other Effects

In addition to the dynamic effects already discussed, there are three significant effects of pipe rupture postulation which have influenced light water reactor plant design. The most significant effect is the requirement for containment buildings which can withstand the internal pressures which would result from a complete loss of primary coolant. Secondly, a pressure wave could travel through a piping system from the point of a break. If this occured in a primary coolant pipe, the result could be a large pressure differential across the reactor core or core barrel. The resulting force could displace those components, resulting in severe damage. This effect is known as "Assymetric Blowdown", and was the original motivation for leak-before-break. A lesser but significant effect combines differential pressurization of a compartment and jet thrust. If a full guillotine break occurred in a primary coolant line near the reactor vessel of primary coolant pump, the thrust of the flashing fluid exiting from the break, along with increased pressure on the side of the break, would exert tremendous loads on the component. Not only are the support requirements for those components greatly increased by break postulation, but in some cases, there has been a requirement to install heaters on the pressure vessel and main coolant supports to ensure that the material remained above the nil-ductility temperature. This would ensure that those supports would deform plastically rather than fracturing in case of an accident.

5.4. Application to Existing Plants

5.4.1. Separation

In nuclear plants which have already been built or are under construction, protection measures taken against dynamic effects of pipe rupture may be eligible for modification through leak-before-break. Other protective measures, specifically those which do not pertain directly to the design of the pipe system are not. Nevertheless, modification in these allowed areas promise substantial improvements to the plant design.

Eliminating physical separation, the most desireable and effective measure, offers substantial savings in pipe cost. Unquestionably, if high energy piping lines were decreased by an average of ten percent in length, millions of dollars could be saved. Reference 33 places the expected savings in the neighborhood of 4.25 million dollars. Whether or not it would be wise to ignore physical separation when laying out reactor piping systems, however, it is unlikely that pipe layout will be altered in existing plants, or in those under construction. To re-route existing pipes would be prohibitive. Even in plants under construction, the additional design effort which would be required would make it unreasonable to change piping system layouts. Although physical separation may lead to longer pipe runs, it is recognized as the one measure which may truly contribute to the safety of the plant (Ref 8). By allowing better access to pipes and other components, and by contributing to piping system flexibility, physical separation is not a likely candidate for elimination from existing plant designs.

5.4.2. Barriers

Physical enclosures around pipes or safety systems could be removed through leak-before-break. Removing these measures would lead to savings in construction, and contribute to easy access to nearby systems. If an enclosure has been built, however, it is probably a concrete wall, which would be difficult to remove. Further, the savings to be gained by removing this requirement from plants which are under construction are uncertain. Since the average plant has only one or two barriers which have been added specifically for pipe rupture protection, and because they are generally set in concrete, removing the requirement for barriers on existing plants would have little effect.

5.4.3. Shields and Restraints

Pipe restraints and jet impingement shields are the two most significant candidates for change due to leak-before-break. Both types of devices are generally bolted in place, and so, may be more easily removed than concrete walls. Some jet impingement protection may still be required because a crack in a high energy pipe could still produce a fluid jet. The shields, however, could be less substantial and fewer, since it would be assumed that the pipe would not move.

Similarly, some pipe restraints may not be removed through leak-before-break. In some instances, pipe restraints are in the form of combination support and restraint. Also, some of the space frame which supports pipe restraints is used for pipe supports. Combination support and restraints are the exception rather than the rule, but most installed space frame structure would not be removed from plants which have already been built, unless the removal were a requirement.

Costs associated with removing rupture restraints include the leak-before-break analysis, and the actual removal. While the cost of justifying leak-before-break has been estimated at approximately $50,000 per system (Reference 34), the cost of

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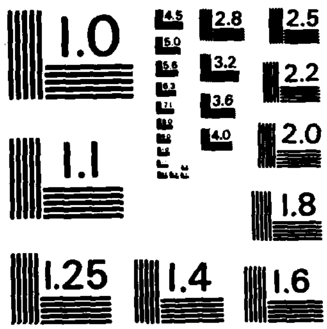

MICROCOPY RESOLUTION TEST CHART
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removing rupture restraints on an optional basis would be nil. If, for example, a group of rupture restraints were removed for in-service inspection during plant shutdown, and re-installation were not required, an initial savings would result from not having to replace the restraints and re-adjust clearances around pipes. While a specific figure for this savings is not available, the estimated cost for initial fabrication and installation of rupture restraints and impingement shields is 7.25 million dollars. If even one tenth of that amount were saved by avoiding re-installation, the potential savings would be $725,000. If the restraints were left off, the plant owners would be free to install regular insulation to the pipes in the vicinity of the former restraints. The normal insulation which is applied to the piping system is much more efficient and inexpensive than the thinner insulation which must be applied in the vicinity of pipe restraints. The thinner insulation is required in order to control the size of the gap which must be left between the pipe and the restraint. Insulation is structurally weak, and is therefore considered part of the gap. The savings in heat loss due to better insulation is estimated to be over 600,000 dollars during the life of a plant (Ref 33). Additional savings would result from a decrease in heat load on the air cooling system, and from better performance of some diagnostic instruments because of the reduced ambient temperature.

There is reason to believe that plant reliability could be significantly improved through removal of pipe restraints. If pipe rupture restraints are installed incorrectly, or if operating conditions vary from those anticipated, it is conceivable that pipes and restraints would come into contact. The reaction forces in this situation would introduce additional stresses into the pipe. In general, higher stresses correspond to higher probabilities for pipe failure. Further, the presence of pipe restraint structure restricts full access to a pipe. This access is needed for proper inspection of the pipes. Both visual and ultrasonic inspection techniques require access to the full circumference of the pipe being inspected. Achieving some degree of access is the reason why restraints must be removed, at least partially, for each inspection. Thus, pipe rupture restraints, which are installed for the purpose of protecting against the effects of pipe break, actually increase the chances of such pipe break.

If one accepts the notion that leaks are precursors to pipe break, then leak detection is an important measure for precluding pipe failure. Many leak detection methods are hindered by pipe restraints. Floor sumps, presently considered to be one of the most reliable leak detection means, require the leaking fluid to reach the floor and flow into a sump. Adding more hardware lengthens the flowpath between the crack and the sump, and allows more places for water to form pools. Hence, both sensitivity and response time are adversely affected. Several proposed leak detection methods would also be improved if pipe restraints were removed. Acoustic emission detection requires transducers which would be mounted on the pipes. The sensitivity would be reduced by the presence of restraints if they contact the pipe and attenuate pipe vibrations. Infrared cameras would be more effective if pipe restraints did not inhibit line-of-sight observation of pipes.

Finally, the fact that restraints would not have to be re-installed and later removed for each inspection would lead to substantial reductions in occupational radiation exposure. This savings alone mitigates any small increased risk of radiation exposure caused by removing the requirement for pipe restraints.

Each cost for removing pipe rupture restraints is outweighed by a benefit. Analysis and removal costs would be outweighed by savings from not having to re-install restraints. Any increased risk of damage due to pipe rupture would be outweighed by the increased pipe system reliability gained by removing restraints. Finally, the savings in radiation dose for workers outweigh the questionable increase in risk due to pipe failure.

5.4.4. Leak Detection.

The sensitivity of installed leak detection instrumentation determines where leak-before-break may be applied. Present leak detection instrumentation is believed to be sensitive to leaks of approximately 1 gallon (3.8 liters) per minute. If a factor of safety of 10 is accepted on this quantity, one could assume that postulated breaks within containment which would leak at least ten gallons (38 liters) per minute from a crack of one half critical size could be excluded using leak-before-break. To exclude consideration of breaks with smaller precursor leak rates, it would be necessary to use leak detection methods with higher sensitivity. It has been suggested that existing methods could be employed differently to achieve higher sensitivities, or new technologies could be employed (Ref 5).

One possible improvement using an existing technology would be to use floor sumps which would draw from a small portion of the containment, such as one steam generator cubicle. Another plan could employ an increased density of thermocouples and pressure transducers to monitor atmospheric conditions inside the containment. New technologies include acoustic emission, which detects ultrasonic vibration of leaking pipes, and infrared television monitors, which would allow operators to see leaks.

Leak detection systems typically consist of measuring devices, connecting cables, some form of data analyzing device, and the output display. All of these could be installed in an existing plant, but it would be more difficult and costly than such installation in a new plant. In many cases, other modifications would be required if a new system were to be installed with increased sensitivity. As an example, floor sumps which cover a small area might require a new floor. Infrared monitors might require the removal of some space frames used for pipe restraint support. Because of the potential difficulties of adding more sensitive leak detection systems to existing plants, it is unlikely that such new systems will be added. Improved calculations or actual testing of leak detection systems might be employed to show that sensitivities are higher than previously stated, or to demonstrate reliability, thus allowing the use of a smaller factor of safety on detectable leak rate. Such measures could be expected to allow exclusion of breaks with associated leak rates of 3-5 gallons (11-19 liters) per minute, but not much lower.

5.4.5. Summary

The applicaton of leak-before-break to existing plants will most likely be confined to the removal of pipe restraints from pipes where the corresponding postulated break is of the order of 5-10 gallons (11-19 liters) per minute. The uncertainties and expected costs associated with installing improved leak detection means rules out expanding the application in existing plants. Because leak-before-break is presently only applicable to dynamic effects of pipe rupture, and protection measures other than restraints are not worth removing, restraint removal is the only significant effect to be expected.

5.5. Application to Future Plants

5.5.1. Compartment Pressurization

Postulated pipe ruptures have significant effects on nuclear plant design beyond those associated directly with pipe system design. These effects stem from three phenomena associated with pipe rupture. The first is containment pressurization, the second is assymetric pressure loads on internal reactor components, and the third is the thrust forces produced at a break as water and steam exit at high velocity.

If a large primary coolant pipe were to break completely, the pressurized water in the primary loop would be free to expand into the reactor containment building. The hot water would, to a large degree, flash to steam as it exited the break site, resulting in a large partial pressure of radioactive steam in the containment building. Because the leak would be so large, it would be nearly impossible to remove the water from the air at a rate comparable to the leak rate.

In order to avoid releasing radioactivity to the atmosphere, containment buildings are designed to contain the increased pressure generated by such an accident. To accomplish this task, containment structures may be prestressed, and a steel liner is added to make it airtight. The access hatches and their seals are machined to provide an airtight seal. The volume of the contanment may also be increased in order to reduce the anticipated pressure from such an accident. Clearly, if containment buildings could be made smaller and of a less complicated design, a major component of the nuclear plant would be much less costly to design and build.

A related issue is compartment pressurization. Any enclosed or partially enclosed area with a pipe subject to rupture in it would be subject to wall loads due to internal pressure generated by escaping steam. In order to avoid destroying the walls of such areas, they must be designed such that they do not restrict the outward flow of air, thus avoiding excessive pressurization. Not only is this additional analysis expensive, but enclosed compartments may be desireable for improved leak detection sensitivity. By limiting the volume of the space which would be sampled, it may be possible to improve the sensitivity to leak rate and to location detection.

5.5.2. Assymetric Blowdown

Assymetric blowdown, as the second effect is known, is the force which would be applied on the reactor core barrel by the pressure wave generated by a sudden primary coolant pipe break. If a guillotine break occurred suddenly enough, the resulting pressure wave would travel up the pipe and into the pressure vessel, displacing the entire core barrel. Experimental results have shown that cracks do not propagate suddenly enough to produce pressure waves of great concern, so this consideration may not be linked to the further application of leak-before-break. Nevertheless, the general application of leak-before-break supports the contention that assymetric blowdown need not be of concern.

5.5.3. Component Supports

The third pipe rupture phenomenon which effects systems other than piping, is jet thrust. Pipe whip is a reaction to the thrust of the fluid exiting from a break. If a break were near the end of a pipe, that same force would act on the component to which the pipe connected. In recognition of this, the supports for the pressure vessel and the primary coolant pumps are designed to withstand a "double-ended guillotine break" in a main coolant line. The result is a set of supports which are much more massive than would otherwise be needed on those components. In fact, because the supports are generally constructed of carbon steel, special heaters have been installed on some primary coolant pump supports to avoid brittle failure in event of a severe accident. These heaters keep those supports above their nil-ductility temperature, so that they would bend, rather than breaking (Ref 30).

Component supports for the primary coolant pumps and the reactor vessel have been strengthened because of pipe rupture consideration. Nevertheless, it is notable that the Livermore study on pipe failure concluded that pipes in a nuclear plant are more likely to fail due to failure of component supports during an earthquake than due to crack growth. In fact, they calculated that pipe breaks due to component support failure are 5 orders of magnitude more likely than breaks due to crack growth. This result suggests that component supports should not be designed to a lesser strength.

The probability of failure for reactor component supports is very low; on the order of 10^{-7}. The Livermore study incorporated conservative assumptions, and even included in its consideration a plant which had not been designed to present day seismic standards. Component supports are designed to remain elastic, even under the combined effects of Safe-Shutdown Earthquake and pipe rupture events. Nevertheless, there is no equivalent to leak detection to warn of growing flaws in component supports during plant operation. In fact, the probability of a large pipe crack and the probability of component support failure are of the save order of magnitude. Further, an SSE event is not the most severe seismic event possible. The important results of the Livermore study are that the probability of failure is small, and that probability is mostly a function of the design method used for the component supports, rather than of the standards to which the supports were designed.

The design criteria associated with compartment and containment pressurization, and with component supports have not been identified by the Nuclear Regulatory Commission as eligible candidates for reconsideration due to leak-before-break criteria, for the time being. This fact is of little consequence for existing reactors, since removing the modificatons associated with these considerations would be impractical. These criteria have not been included as possibilities, however, because that would violate the principle of defense in depth. In-service inspection and leak detection would be used as redundant means to ensure against pipe rupture. If a pipe should break, however, a lack of component support or containment integrities could have severe consequences. The potential for excluding these criteria in the case of future reactors is therefore uncertain. Ongoing studies should contribute to the eventual answer to this question (Ref 19), but other factors, such as how soon future reactors are ordered, will control how much change is made before then.

5.5.4. Dynamic Effects

As in the case of existing plants, future nuclear plants may use leak-before-break to reduce the requirement for pipe rupture restraints and jet impingement shields. In the case of future plants, however, the potential for savings is much greater. Where no requirement for restraints or shields enters into the design, one may save all costs associated with those devices. These costs include structural analysis of the supporting space frame, revised stress analysis of the piping system in the presence of restraints, fabrication and installation of the protective devices, and possible reductions in the length of installed piping. A savings in the cost of financing is expected to dwarf even the above savings.

According to Reference 33, if leak-before-break were applied to a limited number of high energy systems in a future plant, one could expect to save on the order of ninety million dollars, not including any savings associated with improved access to pipes and surrounding areas. The savings expected through leak-before-break are covered in more detail in Appendix D. Using conservative assumptions, the result is that the savings per plant could easily approach $100 million, and would certainly be several tens of millions of dollars.

In any case, the effect of applying leak-before-break to future reactors would certainly include leaving out the restraints and supporting structures for any pipes to which leak-before-break would be applied. Reducing the use of physical separation would be a point of engineering judgement. The resulting freedom from the congestion due to pipe restraint hardware would dramatically change the appearance of the inside of a future nuclear plant.

5.5.5. Costs

The costs of applying leak-before-break to a future plant would include the necessary analysis and material testing, and any increases in inspection or leak detection requirements. These items are analyzed in further detail in Appendix C.

The requirement for improved leak detection which might be required in conjunction with leak-before-break is not clear. There are several reasons for this. One obvious variable is the degree to which leak-before-break would be applied. Another factor would be the type of leak detection employed. Various methods imply different installed and operating costs, and each would have a different reliability. According to Reference 5, an order of magnitude improvement in leak detection sensitivity would cost on the order of hundreds of thousands of dollars, or two orders of magnitude less than the expected savings due to leak-before-break application.

5.5.6. Results

For future nuclear plants, leak-before-break may be applied to systems whose expected leak-before-break leak rate is on the order of tenths of a gallon per minute for medium and large sized (greater than 8 inch or 20.5 cm) systems. The design changes which would be expected might include smaller containment buildings of less substantial construction, and a freedom from the intensive steel space frames which occupy so much of the space inside today's plants. Improved leak detection measures would be installed, both for leak-before-break consideration and for radioactive contamination control.

Chapter 6

RESULTS AND CONCLUSIONS

6.1. Likely Degree of Implementation of Leak-Before-Break

6.1.1. Balanacing Costs And Benefits

Leak-Before-Break should be utilized in cases where the marginal benefits of implementing it outweigh the marginal costs. The tangible and intangible costs associated with implementing leak-before-break are described in appendix C.

The costs of analysis and regulatory approval should vary essentially linearly with the number of systems to which leak-before-break would be applied. No significant economy of scale is expected. Costs for leak detection systems, however, would rise very quickly with the degree of leak-before-break implementation. Leak rates drop quickly with pipe size for small (<8" or 20.5 cm) pipes. More sensitive leak detection would require more sophisticated technologies, system testing, sophisticated analysis of data from detectors, and possibly a higher density of detectors. Thus, the costs of leak-before-break increase with scale.

The marginal benefits of leak-before-break decrease quickly for small pipes, since the restraints on small pipes are themselves small and relatively inexpensive, and the numbers of restraints actually decreases with decreasing pipe size. This is a situation of decreasing returns to scale. As a result, the point where leak-before-break becomes impractical is not very sensitive to uncertainties. Figure 6.1 illustrates how the threshold for applying leak-before-break would be chosen if all of the costs and benefits could be quantified.

6.1.2. Existing plants

Existing nuclear plants, including those under construction, have significant sunk costs and in-place hardware. Analyses have already been accomplished, and any changes allowed would effect mostly the removal of already installed pipe restraints. The exception to this would be that some plants under construction may not have all pipe rupture restraints in place, and their installation might be avoided.

Pipe lines which would be calculated to leak at rates of 10 gallons (38 liters) per minute or greater before reaching one half critical size would be the likely candidates. If normal operating stresses are used to calculate leak rates, then all medium and

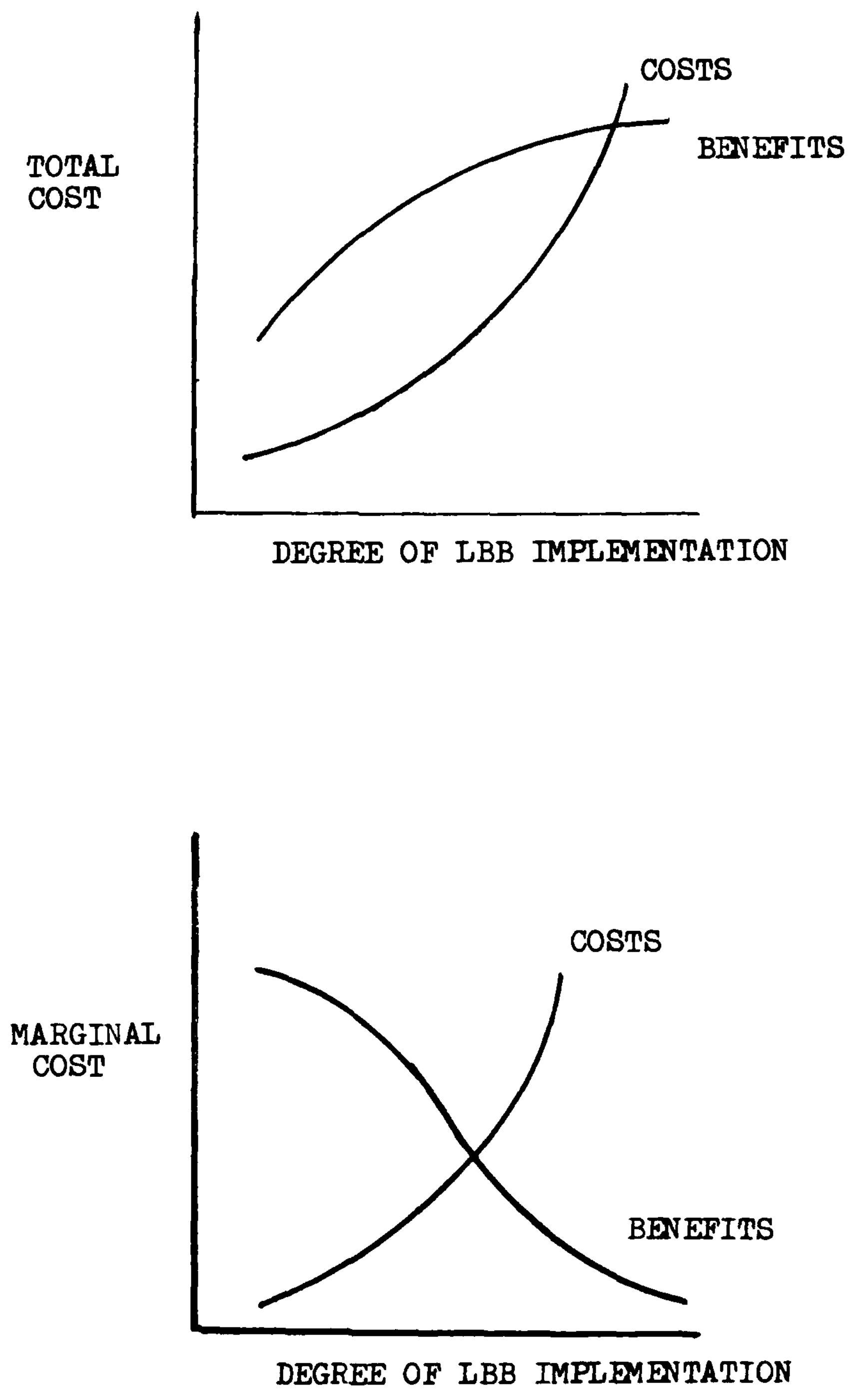

Figure 6.1. Illustration of Cost-Benefit Comparison for LBB

large lines would be expected to qualify for leak-before-break. As a result, the vast majority of pipe restraints could be removed or left out. Any space frame structure, however, would probably be left in place because of the difficulty of removing it, and because of its function of supplying support to pipes and other equipment.

6.1.3. Future plants

Since future plants are those which have not yet been ordered, there is significant flexibility available in applying leak-before-break. Both dynamic effects and other effects of pipe rupture could potentially be excluded from design consideration.

Certainly, rupture restraints would be excluded from the beginning on lines of larger than 8 inch (20.5 cm) nominal diameter. Leak-before-break might even be applied to some groups of six inch (15 cm) pipes, if the additional leak detection means which would be required were justified by the savings in restraint cost. For small lines, leak-before-break will not be applied until significant developments take place in leak detection instrumentation, and the occurrence of high cycle fatigue in small pipes is controlled.

Table 6.1

DESIGN IMPLICATIONS OF LBB

-Leak-Before-Break Originally Intended to
Take Advantage of Installed Leak Detection

-Only Accepted for Eliminating Protection
Against the Dynamic Effects of Pipe Rupture

-Increasing Applicability Requires Improvements in
Leak Detection, With Further Consequences

-Major Changes to be Expected Are
Removal/Modification of
Pipe Rupture Restraints
Jet Impingement Shields
For Systems of Pipes With Nominal Diameters
Greater Than 8 inches or 21 cm.

Other effects of pipe rupture could be excluded from consideration depending on how much time passes before the future plants are designed. In the near future, it is unlikely that the requirement for containment buildings will be reduced. Component supports should, if anything, be fortified, due to their contribution to pipe rupture probability. Table 6.1 summarizes the implications of leak-before-break on PWR design.

6.2. Steps Beyond Leak-Before-Break

6.2.1. Avoiding pipe rupture

Because of reliable inspection techniques and a good understanding of fatigue, it is unlikely that a crack in a nuclear pipe would grow into a break without being detected. The probability is much higher that a pipe would fail because it was hit by something, or because a component support failed.

Leak-before-break originated because leak detection capability is already installed in a nuclear plant. It does not address the conclusion of the Livermore study, and of many experts, that breaks in these pipes are very unlikely. In the future, it is possible that the requirement to consider the dynamic effects of pipe rupture would be removed. Even without leak detection considerations, the plants would be more reliable, economical, and safer. Because the hazard of component support failure is greater than that of crack growth, however, it is also possible that regulation would shift toward requiring more robust designs for these structures.

6.2.2. Leak detection

Under leak-before-break, there is some motivation to improve the installed leak detection capability in a plant. If more sensitive leak detection should result, two side effects are likely. The first is that, unless the detection method is very specific, and able to identify the source and leak rate precisely, reactor operators would be required to shut reactors down more frequently due to detected leaks. Secondly, if some plants install improved leak detection, a rule may evolve which would require the improved instrumentation in all reactors. Indeed, in the present state, the sensitivity and reliability of installed leak detection systems are uncertain.

6.3. Recommendations

6.3.1. Leak detection

Leak detection systems have been installed in reactors for a reason. Leak-before-break adds further purpose. Nevertheless, most detection systems are never tested, and little effort has been applied to producing accurate computations of leak rate sensitivities. Leak detection systems should be tested to verify their sensitivities. If the systems are found to be reliable, then the enormous safety margin applied to leak rates calculated under leak-before-break could be reduced from 10 to some smaller number. If leak detection means are found to be unreliable or inefficient, new detection means, such as infrared and acoustic emission detection should be developed further.

6.3.2. Safety margins

The safety margins chosen for the recommended leak-before-break criteria are very conservative, They are purposely very conservative because of uncertainties in crack behavior and fluid flow prediuncertainties conceing the reliability of leak detection systems. The chosen factors of safety also happen to be just sufficient to allow approval of the Westinghouse request to remove pipe rupture consideration from the primary coolant loops

Table 6.2

RECOMMENDATIONS

- Safety Margins Should Reflect Degree of
  - Confidence in physical models and computations
  - Degree of danger resulting in failure of component

- Need for Protection Measures Against Pipe Rupture Should be Reconsidered
  - Pipe restraints apparently detract from design
  - Low probability of a crack resulting in a break

- Requirements for Leak Detection Should be More Well Defined and Based on
  - Reasons for requiring leak detection
  - Reliability of systems
  - Responses to signals which indicate leaks
  - New leak detection methods

in its plants. With the results of leak detection testing, and evaluations of system reliabilities, some rational choices should be made concerning the appropriate safety margins to use for leak-before-break. Safety margins might be related to the probability of an accident and its consequences, so that more reliable leak detection means or smaller, less menacing pipes would result in lower factors of safety being required.

6.3.3. Employment of Leak-Before-Break

When pipe restraints are removed because the pipe breaks which they protect against would be preceeded by a detectable leak, the operation of the plant presupposes that no unidentified leak of that magnitude exists. It is presently unknown how often leaks of various sizes occur due to various sources. There must be some understanding of how to interpret a detected leak in order to ensure that a leak-before-break leak would cause plant shutdown to avoid pipe rupture, while a non-leak-before-break leak would not. Some information should be gathered concerning the variability of leak rates from sources such as valve packings and pump seals, and some philosophy should be developed concerning when a plant should be shut down for detected leakage.

The development of a leak detection philosophy would require substantial effort. It would require statistical surveys which might show, for example, that a rapidly growing leak is most likely due to a leaky valve packing rather than a pipe crack. Utilities have access to the operational information required, such as leak rate growth versus leak source.

There is presently a justified adversity to new risks and costs in the nuclear industry. To develop this leak detection philosophy would require not only cost, but initiative. Nevertheless, the benefits of improved understanding of plant operation and cost savings make this development worthwhile.

6.4. Summary

Guidelines and appropriate analytical methods exist for applying leak-before-break to systems other than primary coolant loops in light water reactor power plants. The guidelines recommended in NUREG 1061, volume 3 are very conservative, as a first proposal should be. Safety margins which were recommended in that document will be changed to reflect the results of ongoing research and rulemaking.

Elastic Fracture Mechanics, Simplified Homogeneous

Table 6.3

CONCLUSIONS

-Leak-Before-Break May be Applied to Many Piping Systems Without Modifying Leak Detection

-Diminishing Returns Would Result From Applying LBB to Progressively Smaller Systems

-Implementation of LBB in Smaller Systems Will Probably Not Motivate Dramaticaly Improved Leak Detection

-A Majority of Pipe Restraints Could be Removed Through LBB

Equilibrium Model (SHEM), and Limit-Load Analysis are simple models which are applicable to leak-before-break analysis. Because the piping materials used in nuclear plants are generally tough, Limit-Load Analysis applies. Because of the typically subcooled conditions and small openings of any cracks involved, SHEM applies. Finally, because deformations are small in a crack of one half critical size, Linear Elastic Fracture Mechanics will accurately predict crack opening areas. The only cases where these assumptions are marginal are those where A106 steel is used, and where the fluid inside the pipe is not subcooled. The first of these exceptions applies to steam and feedwater lines, while the second applies to the steam lines only. It is notable that many of the postulated breaks which require restraints on lines of these systems lie outside containment, where there are no provisions for leak detection. Applying leak-before-break in these examples could require both additional analysis and leak detection instrumentation.

Leak-Before-Break could achieve several worthwhile objectives if implemented on other pipes. Some likely results of applying leak-before-break include reduced plant cost and complexity, reduced occupational radiation exposure and risk of accidents, increased reliability and availability, and simplified maintenance and inspection.

Leak-before-break should be implemented only in situations where the expected benefits would certainly outweigh the expected costs. The threshold is uncertain, but it is expected to be approximately 6-8 inches (15-20 cm) nominal pipe size. Below this size, great improvements would be required on leak detection sensitivity or reliability, and pipes may be subject to cracking due to high cycle, vibration induced fatigue.

Considering the limited amount of damage expected from the rupture of a small pipe, the required safety margins could be reduced. In fact, the requirement for break postulation should be reconsidered. Since it appears that leak-before-break rules out pipe rupture for large and medium pipes, and small pipes may not be a source of great damage in the unlikely event of their failure, it may someday become possible to eliminate the postulation of pipe ruptures altogether.

APPENDIX A

RECOMMENDED CRITERIA FOR LEAK-BEFORE-BREAK

From chapter 5, NUREG 1061, volume 3.

This section contains the Task Group's recommendations for application of the leak-before-break (LBB) aproach in the NRC licensing process. The LBB approach means the applicaton of fracture mechanics technology to demonstrate that high energy fluid piping is very unlikely to experience double-ended guillotine ruptures or their equibalent as longitudinal or diagonal splits.

The Task Group's recommendations and discussion are founded on current and ongoing NRC staff actions as presented in Appendices A and B (of NUREG 1061, v.3).

Applicants and licensees who choose to justify mechanistically that breaks in high energy fluid system piping need not be postulated should provide submittals that comply with the recommended criteria in this section of the report. As a result of this justification, protection of structures, systems, and components important to safety against the dynamic effects of such postulated ruptures would not be required.

LIMITATIONS

The Task Grop recomends that the following limitations apply to the mechanistic evaluation of pipe breaks in high energy fluid system piping

a. For specifying design criteria for emergency core coooant systems, containments, and other engineered safety features, loss of coolant shall be assumed in accordance with existing regulations, i.e., to be through an opening equivalent to twice the pipe flow area up to and including the largest diameter pipe in the system. The evaluation of environmental effects should be considered on a case-by-case basis.

b. The LBB approach should not be considered applicable to high energy fluid system piping, or portions thereof, that operating experience has indicated particular susceptibility to failure from the effects of corrosion (e.g., intergranular stress corrosion cracking) water hammer or low and high cycle (i.e., thermal, mechanical) fatigue.

c. For plants for which there is an operating license or construction permit, component (e.g., vessels, pumps, valves) and piping support structural integrity should be maintained with no reduction in margin for the Final Safety Analysis Report (FSAR) or

Preliminary Safety Analysis Report (PSAR) loading combination that governs their design.

d. The LBB approach should not be considered applicable if there is a high probability of degradation or failure of the piping from more indirect causes such as fires, missiles, and damage from equipment failures (e.g., cranes), and failures of systems or components in close proximity.

e. The LBB approach should not be considered applicable to high energy piping, or portions thereof, for which verification has not been provided that the requirements of I & E Bulletin 79-14 ("Seismic Analyses for As-Built Safety-Related Piping Systems"), have been met.

f. The LBB approach described in this report is limited in application to piping systems where the material is not susceptible to cleavage-type fracture over the full range of systems operating temperatures where pipe rupture could have significant adverse consequences.

GENERAL TECHNICAL GUIDANCE

To place the above limitations in perspective and to provide guidance to potential users of the LBB approach, each step of the process required to develop the requisite technical justification for a LBB submittal is described in general terms below. A detailed description of the acceptance criteria that is listed below should be used by the staff for evaluation of each submittal follows this general discussion.

a. Provide a discussion to support the conclustion that this piping run or system does not fall within the limitations delineated in Section 5.1 (above).

b. Specify the type and magnitude of the loads applied (forces, bending and torsional moments), their source(s) and method of combinaton. Identify the location(s) at which the highest stresses coincident with poorest material properties occur for base materials, weldments, and safe ends.

c. Identify the types of materials and materials specifications used for base metal, weldments and safe ends, and provide the materials properties including appropriate toughness and tensile data, long-term effects such as thermal aging and other limitations.

d. Postulate a flaw at the location(s) specified in (b) above that would be permitted by the acceptance criteria of Section XI of the ASME Boiler & Pressure Vessel Code. Demonstrate by fatigue crack growth analysis for Code Class I piping that the crack will not grow significantly during service.

e. Postulate a throughwall flaw at the location(s) specified in (b) above. The size of the flaw should be large enough th. , leakage is assured of detection with margin using the installed leak detection capability when the pipes are subjected to normal operating loads. If auxiliary leak detection systems are relied on, they should be described.

f. For geometrically complex lines or systems, performance of a system evaluation should be considered.

g. Assume that a safe shutdown earthquake (SSE) occurs prior to detection of the leak to demonstrate that the postulated leakage flaw is stable under normal operating plus SSE loads for a long period of time, that is, crack growth if any is minimal during an earthquake.

h. Determine flaw size margin by comparing the selected leakage size flaw (Item e) to critical size crack. Using normal plus SSE loads, demonstrate that there is a margin of at least 2 between the leakage size flaw and the critical size crack to account for the uncertainties inherent in the analyses and leak detection capability.

i. Determine margin in terms of applied loads by a crack stability analysis. Demonstrate that the leakage-size cracks will not experience unstable crack growth even if larger loads (at least the 2 times the normal plus SSE loads) are applied. Demonstrate that crack growth is stable and the final crack size is limited such that a double-ended pipe break will not occur.

The piping materials toughness (J-R curves) and tensile (stress-strain curves) properties should be determied at temperatures near the upper range of normal plant operation. The test data should demonstrate ductile behavior at these temperatures.

k. Ideally the J-R curves whould be obtained using specimens whose thickness is equal to or greater than that of the pipe wall. The specimen should be large enough to provide crack extensions up to an amount consistent with the J/T condition determined by analysis for the application. Because practical specimen size limitations exist, the ability to obtain the desired amount of experimental crack extension may be restricted. In this case, extrapolation techniques may be used if appropriate as described in Section A2.4.3 (Appendix A, NUREG 1061, v.3).

l. The stress-strain curves should be obtained over the range from the proportional limit to maximum load.

m. Ideally, the materials tests should be conducted using archival material for the pipe being evaluated. If archival material is not available, tests should be conducted using specimens from three heats of material having the same material specification. Test material should include base and weld metals.

n. At least two stress-strain curves and two J-resistance curves should be developed for each of a minimum of three heats of materials having the same material specifications and thermal and fabrication histories as the in-service piping material. If the data are being developed from an archival heat of material, a minimum of three stress-strain curves and three J-resistancae curves from that one heat of material is sufficient. The tests should be conducted at temperatures near the upper range of normal plant operation (e.g., 550F). Tests should also be conducted at a lower temperature, which may represent a plant condition (e.g., hot standby) where pipe break would present safety concerns similar to normal operation. These tests are intended only to determine if there is any significant dependence of toughness on temperature over the temperature range of interest.

One J-R curve and one stress-strain curve for one base metal and weld metal are considered adequate to determine temperature dependence.

o. As indicated in Section 5.9.1 there are certain limitations that currently preclude gereric use of limit-load analyses to evaluate leak-before-break conditions for eliminating pipe restraints. However, the Task Group believes that limit-load analysis can be used to demonstrate acceptable leak-before-break margins for the application, provided the limit moment is greater than the applied (normal operation plus safe shutdown earthquake (SSE)) moment at any locaton in the pipe run by a factor of at least three. Limit moment should be determined from Eq. (A-19) in Appendix A where the flow stress is determined from ASME Code minimum properties. Data obtained from future tests (see Section 10.0) may provide information that would allow less restrictive use of limit-load analyses for justifying elimination of pipe restraints.

The preceding description of the steps in performing a LBB analysis assumes that circumferentially oriented postulated cracks are limiting. If this is not the case, the analyses described in the above steps should also include the postulation of axial cracks and/or elbow cracks.

```
1 REM **********BREAK FREE***********
2 REM
3 REM ***THIS CODE WILL CALCULATE****
4 REM ***THE CRITICAL CRACK SIZE*****
5 REM ****AND CALCULATE THE FLOW*****
6 REM ***RATE FROM A CRACK OF ONE****
7 REM *******HALF THAT SIZE.*********
8 REM
9 REM ***WRITTEN BY PAUL E. ROEGE****
10 REM ***********JULY 1985***********
15 REM
16 REM
20 DIM Z$(1),NAME$(8),OLD$(8)
21 OPEN #2,8,0,"D:TEST.DAT"
25 PI=3.141
26 Y=0
27 PRINT #2;Y
30 ? "WHAT SYSTEM SHALL WE CALCULATE?"
40 INPUT NAME$
41 IF NAME$=OLD$ THEN 200
45 IF NAME$=" " THEN GO TO 1500
70 ? "WHAT SIZE PIPE?"
80 INPUT NOM
90 ? "ENTER INNER AND OUTER DIAMETERS."
100 INPUT ID,OD
110 R=(ID+OD)/4:TW=(OD-ID)/2
120 ? "ENTER SM, E, AND SF."
130 INPUT SM,YM,SF
135 YM=YM*1000
140 ? "WHAT IS THE INTERNAL PRESSURE? (PSIG)"
150 INPUT P
160 ? "HOW MANY PSI IS THAT SUBCOOLED?"
170 INPUT SC
171 ? "WHAT IS THE SPECIFIC VOLUME? (SAT LIQ)"
172 INPUT VOL
200 ? "WHAT IS THE MAXIMUM BENDING STRESS?"
205 REM ********FIND CRITICAL CRACK SIZE*********
210 INPUT PB:LHS=PI*PB/(2*SF)
220 PM=P*ID/(4000*TW)
225 ? "ENTER MAX EXTERNAL NORMAL STRESS"
226 INPUT ENS:PM=PM+ENS
```

```
230 RF=PI*PM/(2*SF)
240 T=1
250 RHS=2*COS(T/2+RF)-SIN(T)
260 IF RHS>LHS+0.1 THEN T=T+0.01:GO TO 250
270 IF RHS<LHS-0.1 THEN T=T-0.01:GO TO 250
280 IF RHS>LHS+1.0E-03 THEN T=T+1E-04:GO TO 250
290 IF RHS<LHS-1.0E-03 THEN T=T-1E-04:GO TO 250
300 A=T*R/2:LAMBDA=A/(SQR(TW*R))
305 G=0.16*(LAMBDA^4)+(LAMBDA^2)
310 IF LAMBDA>1 THEN G=0.02+0.81*(LAMBDA^2)+0.3*(LAMBDA
^3)+0.03*(LAMBDA^4)
311 PM=PM-ENS
313 ? "WHAT IS THE OPERATING STRESS DUE TO BENDING?":IN
PUT PB
315 ? "WHAT IS THE OPERATING EXTERNAL NORMAL STRESS?":I
NPUT ENS
316 PM=PM+ENS
320 AP=PM*6.282*R*TW*G/YM
326 TP=T/(2*PI)
327 FO=8.6-(13.3*TP)+(24*(TP^2))
328 FT=22.5-(75*TP)+(205.7*(TP^2))-(247.5*(TP^3))+(242*
(TP^4))
329 IT=2*((T/2)^2)*(1+((TP^1.5)*FO)+((TP^3)*FT))
330 IB=(COS(T/4)^2)*IT
340 ? "THE CRITICAL CRACK SIZE FOR THE ";NOM;" INCH ";N
AME$;" LINE IS:";2*R*T;" INCHES."
345 AB=PB*PI*(R^2)*IB/YM
346 PRINT #2;NAME$,NOM,2*R*T,SC
350 AREA=AP
360 ? "FOR CRACK OPENING DUE SOLELY TO LONGITUDINAL FOR
CE:":GOSUB 400
370 AREA=AP+AB
375 ? "PRESS ANY KEY TO CONTINUE.":INPUT Z$
380 ? "FOR CRACK OPENING DUE TO COMBINED LOADING:":GOSU
B 400
385 OLD$=NAME$
390 GO TO 30
400 REM ********CALCULATE LEAK RATE***
410 D=AREA/A
420 FF=(2*CLOG(D/4E-04)+1.74)^-2
430 FRIC=3.68745+(FF*TW/D)
435 IF SC=0 THEN SC=P
440 MF=SQR(9273.6*SC/(VOL*FRIC))
450 VFR=0.05002*MF*AREA
455 PRINT #2;AREA,D,FF,MF,VFR
456 IF SC=P THEN GOSUB 1000:GO TO 510
```

```
460 PRINT "CRACK OPENING AREA:",AREA
470 ? "HYDRAULIC DIAMETER:",D
480 ? "FRICTION FACTOR:",FF
490 ? "MASS FLUX (LBM/FT2-SEC):",MF
500 ? "LEAK RATE (GPM):",VFR
510 RETURN
1000 ? "THIS SYSTEM IS SATURATED."
1020 GMIN=144*0.0165*(P^0.97)
1025 QMIN=0.05002*GMIN*AREA
1030 ? "LEAK RATE IS BETWEEN ";QMIN;" AND ";VFR;" GPM."
1035 PRINT #2;QMIN
1040 RETURN
1500 ? "END OF RUN"
1510 CLOSE #2
```

APPENDIX C

EXPECTED COSTS OF LEAK-BEFORE-BREAK

Applying leak-before-break to a future plant would require the necessary analysis and material testing, and any increases in inspection or leak detection requirements. The first of these items was addressed in Reference 34, the text of which is given below. According to that estimate, the materials testing and fracture mechanics analysis of one system would cost \$59,200. That figure includes \$14,000 for material testing, and \$45,200 for fracture mechanics analysis. The cost of some fatigue analysis and possibly some two-phase flow calculations would increase that amount somewhat. Each of the three analyses could require the application of an iterative computer model. The fracture mechanics analysis is the most difficult. Assuming roughly comparable costs for each of the three analyses, the total cost for justifying leak-before-break in one system would be on the order of \$14,000 + 3 x \$45,200, or roughly \$150,000.

The cost of additional leak detection requirements is less certain. Reference 5 suggests that the cost per detection unit for a localized leak detection scheme would cost of the order of \$100,000. Allowing for several applications of localized leak detection complete with instrumentation, and the required analysis, leak-before-break cost should be of the order of a million dollars.

Some important, though less tangible costs of leak-before-break are also associated with improved leak detection. First, it is likely that improved leak detection would detect more leaks. Unless the reactor operator could be certain that a small, newly detected leak did not correspond to a danger of pipe rupture, he might have to shut down the reactor. Thus, the availability of the plant would decrease whenever a small leak unrelated to pipe rupture is detected, such as from a leaky valve packing. The second danger is that improvements in the state of the art of leak detection would lead to new requirements for installed leak detection capability in all plants. Such increased requirements are likely in light of past experence in nuclear plant regulation.

TEXT OF REFERENCE 31, "ESTIMATED COST OF THE TECHNICAL JUSTIFICATION FOR A LEAK-BEFORE-BREAK SUBMITTAL

By Vladimir Zilberstein, Stone and Webster Engineering Corporation.

An estimated cost of the technical justification for a leak-before-break submittal (per system) is $59,000. This cost is based on $80/hr and an estimated effort in man-hours itemized on the attached copy of section 5.2 (NUREG 1061, vol 3). The total estimated engineering time is 515 man-hours. The total estimated time which includes management and clerical help is 565 man-hours.

The cost of testing is based on an estimated cost of obtaining a set of data consisting of one J-R and one - curve ($1000) and the number of the data sets required by para. 12, p. 5-4 of NUREG 1061, v.3. λ

"two sets for each of a minimum of three heats... at temperatures near the upper range of normal plant operation," and one set "at a lower temperature... (e.g. hot standby)," i.e. a total of seven data sets for base metal and seven data sets for weld metal.

Thus, the estimated cost of testing is $14,000.

The total estimated cost of the justification for leak-before-break submittal (per system) is then

565 x 80 + 14,000 = 45,200 + 14,000 = $59,200

APPENDIX D

ESTIMATED SAVINGS FROM LEAK-BEFORE-BREAK

Figure D.1, from Reference 33, is an estimate of savings to be gained by applying leak-before-break to a light water reactor power plant of current design. One should note that certain assumptions were made in this estimate. The first item to be eliminated, for example is a hazards evaluation. This refers to an evaluation of targets vulnerable to pipe whip. If this analysis must be done to identify locations requiring leak-before-break analyses, then this 3-1/4 million dollar entry would not lead to a net savings. Further, if the estimates involving finance charges were based on a rule of thumb which sets finance charges equal to the actual item cost, then the potential savings are reduced by 6-1/2 million dollars, or more than nine percent. On the other hand, some savings might have been left out. As stated in the figure, no consideration was given to any expected savings in labor costs saved because of easier inspection or maintenance, or to operating costs saved due to reduced heat loss.

Figure D.1

POTENTIAL COST SAVINGS OF LBB CRITERIA FOR FUTURE PLANTS

ASSUMPTIONS

Breaks will continue to be postulated in a limited number of large bore systems (e.g. pressurizer surge line, portions of feedwater, etc.) and in small bore systems (where the burden of justifying LBB may not prove cost effective).

Operating costs associated with hindering ISI (in-service inspection) and maintenance and decommissioning costs have not been considered.

DIRECT COSTS

Hazards Evaluation	
Engineering	3,250,000
Design and Procurement of Restraints, Shields and Supporting Steel	
Engineering	3,250,000
Fabrication and Installation	7,250,000
Piping Restress to Shift Break Points or Lower Stress, Additional Supports Required	
Engineering	500,000
Fabrication and Installation	750,000

Engineering	1,250,000
Reinforcing Bars (Installed)	1,500,000
Improved Piping Layout (Reduced pipe length)	4,250,000
Specialty Insulation	125,000
Primary Equipment Supports	
Engineering	2,750,000
Fabrication and Installation	5,000,000
<u>OPERATING COSTS</u>	
Lost Power and Heat Removal Due to Reduced Insulation	615,000
<u>INDIRECT COSTS</u>	
Improvement in Construction Schedule (30-90 days est.)	30,000,000 (min)
Financing (Interest accrued during construction)	<u>30,000,000</u>
	60,000,000 (min)
Total Savings	90,000,000 (min)

APPENDIX E

SYMBOLS

a	Crack half length
C	Orifice coefficient
D	Hydraulic diameter
E	Young's modulus
f	Friction factor
G	Mass flux
J	J-integral
K	Stress intensity
l	Flow path length
L	Crack length
S	Nominal stress value
T	Tearing modulus
λ	Crack parameter
θ	Crack half angle

APPENDIX F

REFERENCES

1. "ASME Boiler and Pressure Vessel Code, Section III," American Society of Mechanical Engineers

2. Chexal, B., Abdollahian, D., Norris, D., "Analytical Prediction of Single Phase and Two-Phase Flow Through Cracks in Pipes and Tubes," Electric Power Research Institute, Palo Alto, CA

3. Chexal, B., Abdollahian, D., "Calculation of Leak Rates Through Cracks in Pipes and Tubes," Electric Power Research Institute, Report EPRI NP 3395, Palo Alto, CA, January, 1984.

4. Chou, J., Elia, F., "Thermal-Hydraulic Analysis for PWR Coolant Pressure Boundary Leakage Detection System Design," Stone and Webster Engineering Corporation Report B1-42243-5041, Boston, MA, June 1985.

5. Day, B, "A Study of Leak Detection for the Leak-Before-Break Concept," SM Thesis, Dept of Nuc. Eng., M.I.T, to be published.

6. "Design Basis for Protection of Light Water Nuclear Power Plants Against Effects of Postulated Pipe Rupture," American Nuclear Society ANSI/ANS 58.2-1980.

7. "Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping," U.S. Nuclear Regulatory Commission Standard Review Plan 3.6.2.

8. East, G., personal communications, Stone and Webster Engineering Corporation, meetings on various dates.

9. "Environmental and Missile Design Bases," 10CFR Part 50, Appendix A, General Design Criteria 4.

10. Fauske, H., "Flashing Flows - Some Practical Guidelines for Emergency Releases," Fauske and Associates, to be published in Plant/Operations Progress, 1985.

11. "Final Safety Analysis Report of a 4-loop Westinghouse PWR Nuclear Power Station," Stone and Webster Engineering Corporation.

12. Frank, L., et al., "Pipe Cracking Experience in Light Water Reactors," U. S. Nuclear Regulatory Commission Report NUREG 0679, Washington, D. C., August 1980.

13. Giambusso, A., letter to licensees and applicants for operating licenses concerning protection from dynamic effects of pipe rupture, US Atomic Energy Commission, December 1972.

14. Gurnis, G., personal communication, Stone and Webster Engineering Corporation, meeting on May 6, 1985.

15. Holman, G., Chou, C., "Assessment of Value-Impact Associated with the Elimination of Postulated Pipe Ruptures from the Design Basis for Nuclear Power Plants," Lawrence Livermore Laboratory Report UCID 20397, Livermore, CA, March 1985.

16. Holman, G., personal communication, Lawrence Livermore Laboratory, telephone conversation on May 22, 1985.

17. Keenan, J., et al., "Steam Tables," John Wiley and Sons, New York, 1978.

18. Klecker, R. W., "Report of the U. S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks," U. S. Nuclear Regulatory Commission Report NUREG 1061, vol. 3, Washington, D.C., October, 1984.

19. Klecker, R., personal communication, U. S. Nuclear Regulatory Commission, telephone conversation on May 22, 1985.

20. Kumar, V., German, M., Shih, C., General Electric Corporation, "Engineering Approach for Elastic-Plastic Fracture Mechanics," Electric Power

Research Institute Report EPRI NP 1931, Schenectady, NY, July 1981.

21. "Mark's Standard Handbook for Mechanical Engineers, 8th edition", Theodore Baumeister, editor, McGraw Hill, New York, 1978

22. Norris, D., et al., "PICEP: Pipe Crack Evaluation Program," Electric Power Research Insitute Report EPRI NP 3596-SR, Palo Alto, CA, August 1984.

23. Paris, P. C., Tada, H., Del Research Corp, "The Application of Fracture Proof Design Methods Using Tearing Instability Theory," U. S. Nuclear Regulatory Commission Report NUREG/CR 3464, St Louis, MO, September 1983.

24. "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," U. S. Nuclear Regulatory Commission Standard Review Plan 3.6.1, Washington, D. C., July 1981.

25. "Protection Against Pipe Whip Inside Containment," U. S. Atomic Energy Commission Regulatory Guide 1.46, Washington, D. C., May 1973.

26. Sanders, J. L., Jr., "Circumferential Through-Cracks in Cylindrical Shells Under Tension," ASME Journal of Applied Mechanics, Vol. 49, pp. 103-107, 1982.

27. Sanders, J. L., Jr., "Circumferential Through-Cracks in Cylindrical Shells Under Bending," ASME Journal of Applied Mechanics, Vol. 50, No. 1, 1983.

28. Sekerak, P. "Task No. 3: Evaluation of LBB Criteria," Stone and Webster Engineering Corporation report, to be published.

29. "Specification for Piping Engineering and Design, ASME III, Code Classes 1, 2, and 3, ANSI B31.1, Class 4 ", Specification number 2280.000-149, Rev 6, Stone and Webster Engineering Corporation, Boston, December 1984.

30. Stahlkopf, Karl, personal communication, Electric Power Research Institute, letter dated 31 May

1985.

31. "Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection," ISA-S67.03, Instrument Society of America, 1982.

32. Tang, H. T., et al., "Experimental Investigation of High Energy Pipe Leak and Rupture Phenomena," Electric Power Research Institute.

33. Van Sickel, A., correspondence with C.K. Chou regarding Estimated Savings Due to Leak-Before-Break, Stone and Webster Engineering Corporation, letter dated 14 December 1984.

34. Zilberstein, V., Stone and Webster Engineering Corporation, "Estimated Cost of the Technical Justification for a Leak-Before-Break Submittal," memorandum dated April 2, 1985.

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